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ENGINEERING CHANGE NOTICE

Page 1 of 2

1. ECN 671031

Proj. ECN

2. ECN Category (mark one)	3. Originator's Name, Organization, MSIN, and Telephone No.	4. USQ Required?	5. Date
Supplemental ☐ Direct Revision ☒ Change ECN ☐ Temporary ☐ Standby ☐ Supersedure ☐ Cancel/Void ☐	RD Carrell, Site-Wide SNF, L0-30, 521-6421	☒ Yes ☐ No	03/10/02
	6. Project Title/No./Work Order No.: 200 Area Interim Storage Area, W-518	7. Bldg./Sys./Fac. No.: 212-H	8. Approval Designator: SN,Q,E,D
	9. Document Numbers Changed by this ECN (includes sheet no. and rev.): HNF-3553, Rev. 1	10. Related ECN No(s).: NA	11. Related PO No.: NA

12a. Modification Work	12b. Work Package No.	12c. Modification Work Completed	12d. Restored to Original Condition (Temp. or Standby ECNs only)
☐ Yes (fill out Blk. 12b) ☒ No (NA Blks. 12b, 12c, 12d)	W-518	NA Design Authority/Cog. Engineer Signature & Date	NA Design Authority/Cog. Engineer Signature & Date

13a. Description of Change — 13b. Design Baseline Document? ☐ Yes ☒ No

There were 2 significant areas of change related to the LWR fuel storage. New designs for the LWR canister and internal components were completed, and the assumption that the LWR fuel was intact was deleted. New designs were prepared for the LWR inner canister, PWR fuel assembly internal support assembly, rod consolidation container, and rod consolidation container support assembly. The canister now has a shield plug that is welded after loading. The configuration of the support baskets changed, and the rod consolidation container is now a square box rather than a split pipe configuration. System descriptions and figures have been revised to depict these final designs. Figures D2-14 and D2-15 were replaced with the new configurations. The approach to allow damaged fuel in the LWR canisters results in a TSR change, as the previous analyses assumed intact fuel and that assumption was protected in the existing TSR. In addition, the outstanding ABCRs related to the USQE's since the last revision, have been incorporated and this revision is also considered an Annual Update of Annex D.

14a. Justification (mark one)	14b. Justification Details
Criteria Change ☐ Design Improvement ☒ Environmental ☐ Facility Deactivation ☐ As-Found ☐ Facilitate Const. ☐ Const. Error/Omission ☐ Design Error/Omission ☐	New designs were prepared for the LWR inner canister, PWR fuel assembly internal support assembly, rod consolidation container, and rod consolidation container support assembly. The canister now has a shield plug that is welded after loading. The configuration of the support baskets changed, and the rod consolidation container is now a square box rather than a split pipe configuration.

15. Distribution (include name, MSIN, and no. of copies)	RELEASE STAMP
See attached Distribution list.	DATE: JUL 16 2002 STA: 3 HANFORD RELEASE ID: 18

ENGINEERING CHANGE NOTICE

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1. ECN (use no. from pg. 1): ECN 671031

16. Design Verification Required: (●) Yes ( ) No

17. Cost Impact

	ENGINEERING	CONSTRUCTION
Additional	$ NA	$ NA
Savings	$	$

18. Schedule Impact (days): Improvement NA; Delay

19. Change Impact Review: Indicate the related documents (other than the engineering documents identified on Side 1) that will be affected by the change described in Block 13. Enter the affected document number in Block 20.

SDD/DD	☐	Seismic/Stress Analysis	☐	Tank Calibration Manual	☐
Functional Design Criteria	☐	Stress/Design Report	☐	Health Physics Procedure	☐
Operating Specification	☐	Interface Control Drawing	☐	Spares Multiple Unit Listing	☐
Criticality Specification	☐	Calibration Procedure	☐	Test Procedures/Specification	☐
Conceptual Design Report	☐	Installation Procedure	☐	Component Index	☐
Equipment Spec.	☐	Maintenance Procedure	☐	ASME Coded Item	☐
Const. Spec.	☐	Engineering Procedure	☐	Human Factor Consideration	☐
Procurement Spec.	☐	Operating Instruction	☐	Computer Software	☐
Vendor Information	☐	Operating Procedure	☐	Electric Circuit Schedule	☐
OM Manual	☐	Operational Safety Requirement	☐	ICRS Procedure	☐
FSAR/SAR	☐	IEFD Drawing	☐	Process Control Manual/Plan	☐
Safety Equipment List	☐	Cell Arrangement Drawing	☐	Process Flow Chart	☐
Radiation Work Permit	☐	Essential Material Specification	☐	Purchase Requisition	☐
Environmental Impact Statement	☐	Fac. Proc. Samp. Schedule	☐	Tickler File	☐
Environmental Report	☐	Inspection Plan	☐	Tech Safety Req's.	☒
Environmental Permit	☐	Inventory Adjustment Request	☐		☐

20. Other Affected Documents: (NOTE: Documents listed below will not be revised by this ECN.) Signatures below indicate that the signing organization has been notified of other affected documents listed below.

Document Number/Revision	Document Number/Revision	Document Number/Revision
SNF-5047, 200 Area Interim Storage Area Technical Safety Requirements		

21. Approvals

	Signature	Date
Design Authority	DM Johnson	3/13/02
Cog. Eng.	RD Carrell	3/11/02
Cog. Mgr.	RL McCormack	3/11/02
QA	DW Smith	3/13/02
Safety	RM Crawford	03/13/02
Environ.	DJ Watson	3/14/02
Other	AL Ramble	3/13/02
Operations	MS Bush	3/14/02
PRC	OM Serrano (J.J. Klos for)	3-14-02

	Signature	Date
Design Agent		
PE		
QA		
Safety		
Design		
Environ.		
Other		

DEPARTMENT OF ENERGY

Signature or a Control Number that tracks the Approval Signature

Letter #

ADDITIONAL

A-7900-013-3 (10/97)

DISTRIBUTION SHEET

To Distribution	From RD Carrell, Site Wide SNF	Page 1 of 1
Project Title/Work Order Annex D - 200 Area ISA Final SAfety Analysis Report HNF-3553 Rev 2 (W-518)		Date 3/10/02 EDT No. ECN No. 671031

Name	MSIN	Text With All Attach.	Text Only	Attach./ Appendix Only	EDT/ECN Only
G.D. Bazinet	S8-06				x
J.D. Brasfield	S2-45				x
M.S. Busch	S8-03				x
D.M. Johnson	L0-30	x			
R.L. McCormack	L0-30	x			
D.W. Smith	X3-80				x
L.J. Garvin	X3-79				x
R. D. Carrell	L0-30	x			
D. J. Watson	X3-79				x
O. M. Serrano	X3-64				x
A. L. Ramble	A0-26				x
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SNF Project Files	R3-11	x			

A-6000-135 (10/97)

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Annex D - 200 Area Interim Storage Area Final Safety Analysis Report

Prepared for the U.S. Department of Energy
Assistant Secretary for Environmental Management

Project Hanford Management Contractor for the
U.S. Department of Energy under Contract DE-AC06-96RL13200

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Richland, Washington

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Annex D - 200 Area Interim Storage Area Final Safety Analysis Report

Document Type: SAR Division: SNF Project: W-518

Carrell, RD
for Fluor Hanford

Date Published
March 2002
Prepared for the U.S. Department of Energy

Assistant Secretary for Environmental Management

Project Hanford Management Contractor for the
U.S. Department of Energy under Contract DE-AC06-96RL13200

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[signature] 3/14/02
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458 PAGES

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RECORD OF REVISION	(1) Document Number HNF-3553, Annex D	Page 1 of 1

(2) Title

Annex D – 200 Area Interim Storage Area Final Safety Analysis Report

CHANGE CONTROL RECORD			
(3) Revision	(4) Description of Change - Replace, Add, and Delete Pages	Authorized for Release	
		(5) Cog. Engr. Date	(6) Cog. Mgr. Date
0	EDT: 626891	RD Carrell	JD Carlson
1	ECN: 666700, per 10CFR830 annual update: update references, USQ review changes, minor org changes	RD Carrell	RL McCormack
2 RS	ECN 671031 There were 2 significant areas of change related to the LWR fuel storage. New designs for the LWR canister and internal components were completed, and the assumption that the LWR fuel was intact was deleted. New designs were prepared for the LWR inner canister, PWR fuel assembly internal support assembly, rod consolidation container, and rod consolidation container support assembly. The canister now has a shield plug that is welded after loading. The configuration of the support baskets changed, and the rod consolidation container is now a square box rather than a split pipe configuration. System descriptions and figures have been revised to depict these final designs. Figures D2-14 and D2-15 were replaced with the new configurations. The approach to allow damaged fuel in the LWR canisters results in a TSR change, as the previous analyses assumed intact fuel and that assumption was protected in the existing TSR. In addition, the outstanding ABCRs related to the USQEs since the last revision, have been incorporated and this revision is also considered an Annual Update of Annex D.	for RD Carrell 7/9/02	for RM Crawford 7/9/02

VOLUME 5

ANNEX D – 200 AREA INTERIM STORAGE AREA

FINAL SAFETY ANALYSIS REPORT

April 2002

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EXECUTIVE SUMMARY

DE.1 FACILITY BACKGROUND AND MISSION

The 200 Area Interim Storage Area (200 Area ISA) at the Hanford Site provides for the interim storage of non-defense reactor spent nuclear fuel (SNF) housed in aboveground dry cask storage systems. The 200 Area ISA is a relatively simple facility consisting of a boundary fence with gates, perimeter lighting, and concrete and gravel pads on which to place the dry storage casks. The fence supports safeguards and security and establishes a radiation protection buffer zone. The 200 Area ISA is nominally 200,000 ft^2 and is located west of the Canister Storage Building (CSB). Interim storage at the 200 Area ISA is intended for a period of up to 40 years until the materials are shipped off-site to a disposal facility. This Final Safety Analysis Report (FSAR) does not address removal from storage or shipment from the 200 Area ISA.

Three different SNF types contained in three different dry cask storage systems are to be stored at the 200 Area ISA, as follows:

- **Fast Flux Test Facility Fuel** – Fifty-three interim storage casks (ISC), each holding a core component container (CCC), will be used to store the Fast Flux Test Facility (FFTF) SNF currently in the 400 Area.
- **Neutron Radiography Facility (NRF) TRIGA**[1] – One Rad-Vault[2] container will store two DOT-6M[3] containers and six NRF TRIGA casks currently stored in the 400 Area.

[1] TRIGA is a trademark of General Dynamics Corporation.
[2] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.
[3] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.

- **Commercial Light Water Reactor Fuel** – Six International Standards Organization (ISO) containers, each holding a NAC-1 cask[4] with an inner commercial light water reactor (LWR) canister, will be used for commercial LWR SNF from the 300 Area.

An aboveground dry cask storage location is necessary for the spent fuel because the current storage facilities are being shut down and deactivated. The spent fuel is being transferred to interim storage because there is no permanent repository storage currently available.

DE.2 FACILITY OVERVIEW

The 200 Area ISA is located within the Hanford Site 200 East Area and will provide safe outside storage of the SNF, while protecting the fuel through the use of storage systems resistant to man-made and natural phenomena hazards. The majority of the fuel to be stored within the 200 Area ISA will consist of FFTF SNF. However, the 200 Area ISA will also store other SNF from the Hanford Site, including NRF TRIGA fuel from the 400 Area and commercial LWR fuel, currently stored in the 300 Area.

The 200 Area ISA is located a few hundred feet west of the CSB. The footprint of the ISA is nominally 500 ft. by 400 ft., surrounded by a 7-ft. chain-link fence topped with barbed wire. Five gates in the fence control access of vehicles and personnel. Lighting is provided on the perimeter. Within the fenced area are two concrete pads for placement of ISCs and one concrete pad for the ISO containers. The Rad-Vault container will be placed on level gravel.

An equipment storage building (pre-engineered metal building) is proposed near the 200 Area ISA to house lifting devices, impact limiters, transfer casks, and various other types of equipment associated with storage, movement, or transport of casks. This storage building will be located outside the 200 Area ISA fence and provides no safety-related function.

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[4] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

The 200 Area ISA facility components are classified as general service; the TRIGA storage system components are also designated general service, while the FFTF and LWR storage systems are designated safety significant. The CCC and the NAC-1 canister are designated safety class and have both safety-class and safety-significant functions.

No uncontained radioactive materials will be handled at the storage area. Therefore, decontamination and decommissioning efforts should be minimal.

There is interaction of the ISA with existing CSB facilities for surveillance activities.

DE.3 FACILITY HAZARD CLASSIFICATION

A final hazards categorization of the 200 Area ISA facility was performed based on the final hazard analysis (SNF-4820[5]) and accident analyses documentation (Chapter D3.0) for the facility. Consistent with DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*,[6] the final categorization was based on the material-at-risk quantities identified in an individual cask inventory. The ISA material-at-risk quantities were compared against the DOE-STD-1027-92 threshold quantities. The ISA facility final hazard categorization found the ISA facility to be a Hazard Category 2 facility. This categorization level is consistent with the bases and guidance described in DOE-STD-1027-92. Hypothetical release source terms were developed for each fuel type. The 200 Area ISA inventory assumed for the analysis is based on a per cask basis and either includes seven FFTF driver assemblies, all 101 TRIGA elements, or one LWR assembly.

[5] SNF-4820, 1999, *200 Area Interim Storage Area Final Hazard Analysis Report*, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

[6] DOE-STD-1027-92 (Change Notice 1-1997), 1992, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, U.S. Department of Energy, Washington, D.C.

DE.4 SAFETY ANALYSIS OVERVIEW

The 200 Area ISA is a passive storage facility designed to hold up to 70 storage containers of SNF for up to 40 years. Potential hazards associated with receipt handling and storage of these containers are the release of radioactivity, loss of shielding, or loss of configuration control for criticality. Seven design basis accidents (DBAs) were identified and analyzed to evaluate the potential consequences of these hazards to on-site workers and the public. The seven DBAs analyzed are as follows:

- Cask handling/drop
- Mobile crane mechanical failure
- Cask tip over
- Fuel rod rupture
- Seismic event
- Tornado/wind
- Fire.

In all but one case, the consequences for all three fuel types were prevented by the passive design features of the storage systems. In the case of the mobile crane mechanical failure, credit was not taken for the integrity of the NRF TRIGA storage systems and an unmitigated release was assumed. The consequences for this accident and fuel type were found to be significantly below both the off-site limits and the on-site guidelines.

Preventive and mitigative aspects relied on in the accident analyses are shown in Table DE-1.

Table DE-1. Structure, System, and Component Safety Classification Summary.

Fuel type	Structures, systems, and components	Design basis accident analysis	Confinement boundary	Criticality prevention	NRC important-to-safety classification	FSAR section
FFTF	FFTF ISC	SS	X		B	D4.4.1
	CCC	SC [a]		X	A	D4.3.1
NRF TRIGA[b]	Rad-Vault[c]	GS			B	D4.4.2
	NRF TRIGA[b] Cask	GS	X		B	D4.4.3
	DOT-6M[d] Container	GS			B	D4.4.4
	2R Container	GS	X		B	D4.4.4
LWR	ISO Container	SS			B	D4.4.5
	NAC-1 Cask[e]	SS			B	D4.4.5
	LWR Canister	SS/SC [a]	X	X	A	D4.3.2
Not a fuel	Crane	GS [f]			B	Not applicable
Not a fuel	Lifting rigging	GS [f]			B	Not applicable

[a] This item is designated as Safety Class for structural integrity to provide criticality geometry control.

[b] TRIGA is a trademark of General Dynamics Corporation.

[c] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.

[d] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.

[e] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

[f] Critical lift requirements imposed using DOE/RL-92-36, 1993, *Hanford Site Hoisting and Rigging Manual*, accomplish NRC equivalency for important-to-safety Category B.

CCC = core component container.
FFTF = Fast Flux Test Facility.
GS = general service.
ISC = interim storage cask.
ISO = International Standards Organization.
LWR = light water reactor.
NRC = U.S. Nuclear Regulatory Commission.
NRF = Neutron Radiography Facility.
SAR = safety analysis report.
SC = safety class.
SS = safety significant.

DE.5 ORGANIZATIONS

Fluor Hanford is responsible to the DOE for planning, integrating, and managing SNF Project activities, including programs, projects, and operations. Organizational responsibilities related to the SNF Project are summarized in the executive summary, and described in detail in Chapter 17.0 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1.[7]

DE.6 SAFETY ANALYSIS CONCLUSIONS

This FSAR Annex concludes that the 200 Area ISA can be operated safely without exceeding the off-site limits or on-site guidelines.

DE.7 FINAL SAFETY ANALYSIS REPORT ORGANIZATION

The 200 Area ISA FSAR is based on the format and content guidance of DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*,[8] the requirements of DOE Order 5480.23, *Nuclear Safety Analysis Reports*,[9] and complies with Title 10, *Code of Federal Regulations*, Part 830, "Nuclear Safety Management."[10] This report also includes content guidance from NRC Regulatory Guide 3.26, *Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants*,[11] as a result of the DOE regulatory policy described in HNF-SD-SNF-DB-003, *Spent*

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[7] HNF-3553, 2000, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Vol. 1, Rev. 0-B, Fluor Hanford, Incorporated, Richland, Washington.

[8] 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.

[9] DOE Order 5480.23, 1992, *Nuclear Safety Analysis Reports*, U.S. Department of Energy, Washington, D.C.

[10] 10 CFR 830, "Nuclear Safety Management," *Code of Federal Regulations*, as amended.

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

Nuclear Fuel Project Path Forward Additional NRC Requirements.[12] Due to the similar mission of the ISA and the CSB for SNF dry storage, HNF-SD-SNF-SP-012, *Additional Guidance for Including Nuclear Safety Equivalency in the Canister Storage Building and Cold Vacuum Drying Facility Final Safety Analysis Report,*[13] supplemental requirements for the CSB (Table 3) were used for guidance in the preparation of SNF-3446, *Spent Nuclear Fuel Project - Criteria Document Spent Nuclear Fuel Final Safety Analysis Report,*[14] Rev. 1, Appendix E, "FSAR Format and Content – Volume 5: Annex D, 200 Area Interim Storage Area FSAR." Annex D was written in accordance with the criteria document (SNF-3446).

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[12] HNF-SD-SNF-DB-003, 1998, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements,* Rev. 4-A, Fluor Daniel Northwest, Incorporated, Richland, Washington.

[13] HNF-SD-SNF-SP-012, 1997, *Additional Guidance for Including Nuclear Safety Equivalency in the Canister Storage Building and Cold Vacuum Drying Facility Final Safety Analysis Report,* Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

[14] SNF-3446, 2000, *Spent Nuclear Fuel Project - Criteria Document Spent Nuclear Fuel Final Safety Analysis Report,* Rev. 2-A, Fluor Hanford, Incorporated, Richland, Washington.

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CONTENTS

LIST OF TERMS

AC	Administrative Control
ACI	American Concrete Institute
AFFRI	Armed Forces Fuel Research Institute
AL	aluminum
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATM	approved testing material
AWG	American wire gauge
BDBA	beyond design basis accident
BED	building emergency director
BWR	boiling water reactor
CCC	core component container
CE	Combustion Engineering, Incorporated
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
CSB	Canister Storage Building
CSER	criticality safety evaluation report
D&D	decontamination and decommissioning
DBA	design basis accident
DBE	design basis earthquake
DBF	design basis fire
DBT	design basis tornado
DFA	driver fuel assemblies
DOE	U.S. Department of Energy
DORF	Diamond Ordnance Reactor Facility
EPZ	emergency planning zone
FFCR	fuel follower control rod
FFTF	Fast Flux Test Facility
FHA	fire hazard analysis
FSAR	final safety analysis report
GA	General Atomics Corporation
GE	General Electric Company
GRC	galvanized rigid conduit
GS	general service
HWVP	Hanford Waste Vitrification Plant
ID	inner dimension
IEM	interim examination and maintenance
ISA	interim storage area
ISC	interim storage cask

CONTENTS (continued)

ISO	International Standards Organization
ITS	important to safety
LCO	Limiting Condition for Operation
LWR	light water reactor
MCO	Multi-Canister Overpack
MLRS	Multiple Launch Rocket System
MOX	mixed oxide
MPFL	maximum possible fire loss
MT	metric ton
MTHM	metric ton of heavy metal
MWd	megawatt day
NEMA	National Electrical Manufacturers Association
NFPA	National Fire Protection Association
NFS	Nuclear Fuel Services
NPH	natural phenomena hazard
NRC	U.S. Nuclear Regulatory Commission
NRF	Neutron Radiography Facility
NSNF	National Spent Nuclear Fuel
NTS	Nevada Test Site
OCRWM	Office of Civilian Radioactive Waste Management
OD	outer dimension
PFP	Plutonium Finishing Plant
PNNL	Pacific Northwest National Laboratory
PTFE	polytetrafluoroethelene
PUREX	Plutonium-Uranium Extraction (Facility)
PVC	polyvinyl chloride
PWR	pressurized water reactor
QARD	quality assurance requirements and description
RCT	radiological control technician
REDOX	Reduction-Oxidation (Facility)
RL	U.S. Department of Energy, Richland Operations Office
SAR	safety analysis report
SC	safety class
SCBA	self-contained breathing apparatus
SNF	spent nuclear fuel
SRP	segmented rod program
SS	safety significant
SS	stainless steel
SSC	structure, system, and component
TSR	Technical Safety Requirement
UBC	Uniform Building Code

CONTENTS (continued)

WE	Westinghouse Electric Corporation
WESF	Waste Encapsulation and Storage Facility
ZPA	zero-period acceleration

CHAPTER D1.0

SITE CHARACTERISTICS

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CONTENTS

CONTENTS (continued)

LIST OF FIGURES

LIST OF TABLES

LIST OF TERMS

ASCE	American Society of Civil Engineers
CFR	Code of Federal Regulations
CSB	Canister Storage Building
DOE	U.S. Department of Energy
FFTF	Fast Flux Test Facility
FSAR	final safety analysis report
HWVP	Hanford Waste Vitrification Plant
ISA	interim storage area
MCO	Multi-Canister Overpack
MLRS	Multiple Launch Rocket System
NRC	U.S. Nuclear Regulatory Commission
PFP	Plutonium Finishing Plant
PUREX	Plutonium-Uranium Extraction (Facility)
REDOX	Reduction-Oxidation (Facility)
SNF	spent nuclear fuel
SSC	structure, system, and component
WESF	Waste Encapsulation and Storage Facility

D1.0 SITE CHARACTERISTICS

D1.1 INTRODUCTION

The 200 Area Interim Storage Area (ISA) is in the 200 East Area of the U.S. Department of Energy (DOE) Hanford Site immediately to the west of the Canister Storage Building (CSB), as shown in Figure D1-1. The ISA will be used to provide interim storage of non-defense reactor spent nuclear fuel (SNF), as described in DOE/EA-1185, *Environmental Assessment, Management, of Hanford Site Non-Defense Production Reactor Fuel, Hanford Site, Richland, Washington.*

The objective of this chapter is to describe the characteristics of the site on which the ISA is located. This chapter and Chapter 1.0 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1, contain information about regional and Hanford Site characteristics. A detailed description of the ISA structure and storage system is provided in Chapter D2.0. Chapter D1.0 supports the hazard analysis and accident analyses in Chapter D3.0.

D1.2 REQUIREMENTS

The requirements that establish the basis for ISA siting are identified in this chapter and in Section 1.2 of HNF-3553, Volume 1. A discussion of U.S. Nuclear Regulatory Commission (NRC) equivalency requirements is also provided in Section 1.2 of HNF-3553, Volume 1.

In addition to the pertinent requirements identified in Section 1.2 of HNF-3553, Volume 1, the following industry standards are applicable to the ISA safety basis:

- American Society of Civil Engineers (ASCE) 7-93, *Minimum Design Loads for Buildings and Other Structures*
- ASCE Standard 4, *Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures.*

D1.3 SITE DESCRIPTION

A description of the Hanford Site and associated areas is provided in Section 1.3 of HNF-3553, Volume 1. The following sections address the geography, demography, and regional land and water use of the area encompassed by and surrounding the CSB and the adjacent ISA. Siting evaluations performed for the CSB are applicable to the adjacent ISA.

D1.3.1 Geography

D1.3.1.1 Hanford Site Vegetation

Information applicable to all SNF Project facilities is provided in Section 1.3.1.1 of HNF-3553, Volume 1.

D1.3.1.2 Hanford Site Facilities

Information applicable to all SNF Project facilities is provided in Section 1.3.1.2 of HNF-3553, Volume 1.

D1.3.1.3 Boundaries for Evaluation of Accident and Effluent Release Limits

The ISA is within the DOE-controlled zone. Consequences of accident releases from the ISA to collocated workers are calculated in Section D3.4.2 at 100 m from the point of release, in accordance with approved procedures. Routine and accidental releases to the public (offsite receptor) are calculated at the Site boundary shown in Figure D1-2. This is also the location of the controlled area boundary, as the term is defined in Title 10, *Code of Federal Regulations*, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR 72), Section 106, "Controlled Area of an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage."

Radiological consequences calculated in Section D3.4.2 were developed considering only the current Hanford Site boundary. Consequences for a receptor located on Highway 240 were calculated for information purposes only (Scott 1995). DOE and its contractors can control access during emergency and accident conditions. This access control meets the requirements of 10 CFR 72.106(b). Before changes to the Site boundaries are placed into effect by the proper authorities, the calculations in this Final Safety Analysis Report (FSAR) Annex will be reviewed (and reanalyzed if required) in accordance with DOE Order 5480.21, *Unreviewed Safety Questions*.

D1.3.2 Demography

A description of the demography surrounding the Hanford Site is provided in Section 1.3.2 of HNF-3553, Volume 1. Figure D1-3 shows the 1996 200 East Area onsite employee population.

D1.4 ENVIRONMENTAL DESCRIPTION

D1.4.1 Meteorology

Meteorology information applicable to all SNF Project facilities in provided in Section 1.4.1 of HNF-3553, Volume 1.

The specific air transport factors used for accident consequence analyses, and the basis for their calculation and use, are discussed in further detail in HNF-SD-SNF-TI-059, *A Discussion on the Methodology for Calculating Radiological and Toxicological Consequences for the Spent Nuclear Fuel Project at the Hanford Site*, and in Section D3.4.1.2. Data is provided for ground-level releases. To support the accident analyses of Section D3.4.2, air transport factors (χ/Q and χ/Q'), representing the dilution of a contaminant by atmospheric turbulence and diffusion as the contaminant travels downwind, were calculated (HNF-SD-SNF-TI-059). The symbol χ/Q' is the ratio of the average air concentration at the receptor to the average release rate at the release point. This ratio is used to assess potential radiological dose and noncorrosive chemical concentration at downwind locations. The symbol χ/Q is the normalized peak air concentration at the center of a puff, divided by the quantity released, and is used to assess the consequences to a receptor for corrosive chemicals. The χ/Q's for the analyses were calculated using joint frequency distribution data so as to be exceeded only 0.5% of the time (99.5th percentile) for each sector, or to be exceeded only 5% of the time (95th percentile) for data from all sectors combined (the greater of the two calculated values is used in the analyses).

The GXQ computer code, Version 4.0 (WHC-SD-GN-SWD-30002), was used to generate χ/Q and χ/Q' values. GXQ incorporates the methods described in NRC Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*.

The wind rose data (Figure D1-4 and Figure D1-5) indicate that winds from the west-northwest sector occur most frequently (nearly 20% of the time). That is, the emissions are transported toward the east-southeast sector. Winds out of the northwest and west also occur with a relatively high frequency (12% and 11%, respectively). The information provided in these figures was obtained from data presented in HNF-SD-SNF-TI-059.

D1.4.2 Hydrology

Hanford Site hydrology information that is applicable to all SNF Project facilities is provided in Section 1.4.2 of HNF-3553, Volume 1.

D1.4.2.1 Surface Water

Surface water information is provided in Section 1.4.2.1 of HNF-3553, Volume 1.

D1.4.2.2 Vadose Zone

A definition of the vadose zone is provided in Section 1.4.2.2 of HNF-3553, Volume 1.

D1.4.2.3 Aquifers

The 200 Area aquifer systems are discussed in Section 1.4.2.3 of HNF-3553, Volume 1.

D1.4.3 Geology

A description of the Hanford Site geology that applies to all SNF Project facilities is provided in Section 1.4.3 of HNF-3553, Volume 1.

D1.4.3.1 Physiographic Setting of the Hanford Site

Information on the physiographic characteristics applicable to the ISA is provided Section 1.4.3.1 of HNF-3553, Volume 1.

D1.4.3.2 Stratigraphy

Information on the stratigraphy of the Pacific Northwest and the Hanford Site that is applicable to the ISA is provided in Section 1.4.3.2 of HNF-3553, Volume 1.

D1.4.3.3 History of Cataclysmic Flooding in the Pasco Basin

Information on geologic structures that is applicable to the ISA is provided in Section 1.4.3.3 of HNF-3553, Volume 1.

D1.4.3.4 Geologic Structures of the Columbia Basin and Hanford Site

Information on geologic structures that is applicable to the ISA is provided in Section 1.4.3.4 of HNF-3553, Volume 1.

D1.4.3.5 Geology of the Interim Storage Area

The geology of the 200 East Area is summarized in Figure D1-6 through Figure D1-8. The suprabasalt sediments consist of the Ringold formation and Hanford formation. The Ringold formation conforms to the basalt bedrock surface and tilts southeast toward the axis of the Cold Creek syncline. The Ringold formation is dominated by gravel units E and A, which are separated by the Lower Mud unit. These are the main unconfined aquifers. The Ringold formation thins from 164 ft at the south end to nearly pinching out at the north end. The beds have been truncated and are unconformably overlain by the Hanford formation. The Hanford formation is mainly sands and gravelly sands that are between 246 and 328 ft thick at the site. Additional information on the geology at the ISA site is provided in Section 1.4.3 of HNF-3553, Volume 1.

D1.4.3.6 Tectonic Development of the Hanford Site

Information on geologic structures and faults that relate to the Hanford Site seismic hazard analysis that is applicable to the ISA is provided in Section 1.4.3.6 of HNF-3553, Volume 1. Additional information is provided in WHC-SD-W236A-TI-002, *Probabilistic Seismic Hazard Analysis, DOE Hanford Site, Washington.*

D1.4.3.7 Contemporary Stress and Strain

Information on earthquake activity, contemporary stress measurements, and subsidence history that is applicable to the ISA (WHC-SD-W236A-TI-002) is provided in Section 1.4.3.7 of HNF-3553, Volume 1.

D1.4.3.8 Geologic Hazards

D1.4.3.8.1 Seismic Hazard Assessment. The ISA is a Hazard Category 2 facility, as stated in HNF-1755, *Initial Hazard Classification for the 200 East Area Interim Storage Area Project W-518*. Performance Category 3 is applicable for a Hazard Category 2 facility. Per WHC-SD-SNF-DB-009, *Canister Storage Building Natural Phenomena Hazards*, the seismic loads for the ISA Performance Category 3 structures, systems, and components (SSCs) follow the DOE design requirements specified in HNF-PRO-097, *Engineering Design and Evaluation (Natural Phenomena Hazard)*, which implements DOE Order 6430.1A, *General Design Criteria*, Section 1300-3, "Safety Class Criteria." This approach is consistent with the application of seismic criteria within the SNF Project. The Performance Category 3 horizontal and vertical design response spectra values were taken from Table 6, "Design Response Spectra for Performance Category 3 in the 100 and 200 Areas" of HNF-PRO-097. The values for various damping levels umbrella the 100 Area, 200 West Area, and 200 East Area and are slightly higher than 200 East Area-specific values (0.26 *g* rather than 0.24 *g*).

D1.4.3.8.2 Volcanic Hazard Assessment. Information about volcanic hazards applicable to all SNF Project facilities is provided in Section 1.4.3.8.2 of HNF-3553, Volume 1.

D1.4.3.8.3 Subsurface Stability. The ISA is constructed in flood sediments, the youngest sediments being approximately 13,000 years old. There are no areas of potential surface or subsurface subsidence, uplift, or collapse except for the low geologic deformation discussed in Section 1.4.3.6 of HNF-3553, Volume 1. With the exception of the loose, surficial, wind-deposited silt, soils are competent and form good foundations. Several geotechnical studies have been completed in and around existing tank farms. Liquefaction of soils beneath the tank farms is not a credible hazard because the water table is greater than 215 ft below ground surface. Liquefaction cannot occur in dry sediments. Liquefaction is also not a concern because of the results of evaluations discussed below for the Hanford Waste Vitrification Plant (HWVP), which is in close proximity to the ISA.

A geotechnical investigation was performed at the site in 1989 to evaluate subsurface conditions for the design and construction of the HWVP. The results of that investigation are provided in Report 10805-385-016, *Report of Geotechnical Investigations for the Proposed Hanford Waste Vitrification Plant, Hanford Washington*. The HWVP Project was canceled and the site is being used for the CSB. Subsurface conditions were investigated by 17 borings ranging in depth from 20 to 100 ft at the site and in the surrounding area. Geophysical tests, including a series of downhole seismic tests, were conducted in boring VP-8, approximately 400 ft from the ISA/CSB. Borehole VP-15 (Figure D1-9) is at the ISA/CSB site.

The 17 geotechnical investigation borings were advanced without drilling fluid by driving a 6-in. diameter, steel "core barrel" with a "split jar" downhole hammer. As the borings were advanced, an 8-in. diameter steel casing was driven to prevent caving of soils above the sampling depth. Water was occasionally poured into the hole to prevent caving of the loose, dry soils and to aid in recovering samples. Slightly disturbed soil samples were taken at 5-ft intervals by driving a sampler using either a 705-lb hammer falling through a distance of 18 in. or a 825-lb hammer falling through a distance of 28 in. During the sampling process, resistance to penetration of the sampler, in blows per foot, was recorded.

The soil profile in the upper 100 ft consists essentially of the three strata shown in Figure D1-10. Borehole VP-14 is approximately 230 ft west of the ISA/CSB. The stratigraphy in VP-15 (Figure D1-9) is the same as at VP-14. The dynamic soil properties of the site are summarized in Figure D1-11. More detailed descriptions of the geotechnical investigations and interpretations are found in Report 10805-385-016 and in Section 1.4.3.8.2 of HNF-3553, Volume 1.

D1.5 NATURAL PHENOMENA THREATS

Section 2.3 of HNF-2524, *200 East Area Interim Storage Area Preliminary Safety Evaluation*, states that recommendations for the classification of safety-class and safety-significant SSCs are derived from DOE Order 6430.1A and HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements*. Design requirements for natural phenomena hazards are established in Engineering Change Notice 643545 to WHC-SD-SNF-DB-009 and Section 2.3.2 of HNF-2524. A listing of safety-class and safety-significant SSCs required for the ISA are presented in Table D4-2. Table D1-1 presents a summary of natural phenomena design loads and criteria for ISA safety-class SSCs.

D1.6 EXTERNAL HUMAN-GENERATED THREATS

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

D1.6.1 Aircraft Activity

The methodology used in aircraft activity analysis is provided in Section 1.6.1 of HNF-3553, Volume 1. Nine active airports are located within a 24-mi radius of the ISA according to HNF-1786, *Assessment of Aircraft Impact Frequency for the 200 Area Interim Storage Area*. Eight of these are small airports that serve only general aviation aircraft. The Richland Airport, 21 mi southeast of the ISA, supports primarily general aviation operations, but two commercial freight carriers per day land and take off from the airport's runways. The nearest airport with significant commercial and military air activity is the Tri-Cities Airport, 29 mi southeast of the ISA.

Table D1-1. Interim Storage Area Safety-Class Natural Phenomena Design Loads.

Hazard	Load	Design guidance
Seismic	Median response spectra: [a] 0.26 *g* horizontal 0.17 *g* vertical	DOE Order 5480.28 [b] DOE-STD-1020-94 [c] HNF-SD-SNF-DB-009, App C
Straight wind	80 mi/h, fastest mile at 30 ft	ASCE 7-93 [d] DOE-STD-1020-94 [c] (including missiles)
Tornado	Wind speeds 200 mi/h total 160 mi/h rotational 40 mi/h translational	U.S. Nuclear Regulatory Commission NUREG-0800 [e] 3.3.2 Tornado Loading
Volcanic ash	24 lb/ft^2 ground ash load	U.S. Nuclear Regulatory Commission NUREG-0800 [e] 3.8.4 Other Seismic Category Structures
Flooding	Dry site for river flooding Site drainage basin: 7.4 in. for 6-hour probable maximum precipitation Site drainage: 9.2 in. for 6-hour probable maximum precipitation	ANSI/ANS-2.8-1992 [f] U.S. Nuclear Regulatory Commission NUREG-0800 [e] 2.4.2 Floods
Lightning	Lightning protection shall be provided for facility	NFPA 780 [g]
Snow	20 lb/ft^2 ground load	ASCE 7-93 [d]

[a] HNF-PRO-097, 2000, *Engineering Design and Evaluation (Natural Phenomena Hazard)*, Rev. 1, Fluor Hanford, Incorporated, Richland, Washington.

[b] DOE Order 5480.28, 1993, *Natural Phenomena Hazards Mitigation*, U.S. Department of Energy, Washington, D.C.

[c] DOE-STD-1020-94 (Change Notice 1-1996), 1994, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*, U.S. Department of Energy, Washington, D.C.

[d] ASCE 7-93, 1993, *Minimum Design Loads for Building and Other Structures*, American Society of Civil Engineers, New York, New York.

[e] NUREG-0800, 1981, *Standard Review Plan*, U.S. Nuclear Regulatory Commission, Washington, D.C.

[f] ANSI/ANS-2.8-1992, 1992, *Determining Design Basis Flooding at Power Reactor Sites*, American Nuclear Society, La Grange Park, Illinois.

[g] NFPA 780, 1995, *Lightning Protection Systems*, National Fire Protection Association, Quincy, Massachusetts.

The major contributor to the frequency of aircraft impact into the ISA is general aviation aircraft during in-flight operations. The overall frequency of aircraft impact from all sources is 3.55 x 10^{-7}/yr (HNF-1786). DOE-STD-3014-96, *Accident Analysis for Aircraft Crash into Hazardous Facilities*, specifies that if the total impact frequency, calculated according to the method it gives, is less than 10^{-6}/yr conservatively calculated, or 10^{-7}/yr realistically calculated, the safety risk is below the level of concern and no design basis accident analysis is required.

Information on rotary wing aircraft is provided in Section 1.6.1 of HNF-3553, Volume 1. Medical evacuation helicopters' closest approach to the ISA will be a landing pad to the west of the 2704 Building, which is about 0.9 mi from the ISA.

D1.6.2 Other Transportation Accidents

A discussion regarding guidance for evaluating transportation accidents is provided in Section 1.6.2 of HNF-3553, Volume 1.

Highway 240 is 5 mi from the ISA. The nearest railroad not controlled by DOE is at the 1100 Area, located approximately 20 mi south of the ISA. DOE currently owns and controls the railroad north of this point, as discussed in Section 1.3.1 of HNF-3553, Volume 1. At these distances, explosive shipments on roads and railroads not controlled by DOE do not represent a threat to the ISA.

The main roadway and railroad controlled by DOE that pass nearest to the ISA are Route 4, which passes 1,155 ft to the west, and the mainline of the Hanford Railroad, which passes 9,000 ft to the north. The mainline of the Hanford Railroad is outside the above-listed safe distance, but Route 4 is not.

NRC Regulatory Guide 1.91, *Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants*, allows for a risk assessment when the safe distance criterion is not met. If the exposure rate is less than 10^{-7} events/yr by best estimate analysis or less than 10^{-6} events/yr by conservative analysis, then the risk of damage being caused by explosion is sufficiently low. The exposure rate, r, is defined as follows:

$$r = n \times f \times s$$

where

n = explosion rate for the substance and transportation mode in question, per kilometer

f = frequency of shipment for the substance in question in shipments per year

s = exposure distance in km. This is the length of the roadway that would be within the safe distance of any portion of the facility. It is a function of the building dimension parallel to the roadway, the setback of the building from the roadway, and the calculated safe distance, R.

H&R 522-1, *Recommended Onsite Transportation Risk Management Methodology*, concludes that a Hanford Site truck accident rate of 5.5 x 10^{-8} accident/mi should be used for risk analysis. This study was performed for shipment of radioactive materials and took credit for risk reduction factors that would be associated with such shipments. These included driver participation in special safety programs, the road conditions north of the Hanford Site Wye Barricade, and making shipments during off-peak traffic hours. A risk reduction factor of 40 was obtained for Hanford Site two-lane roads such as Route 4 (H&R 522-1). This risk reduction factor also would be appropriate for shipment of large quantities of explosive materials. For this analysis, it is conservatively assumed that the conditional probability that an explosion will result as a consequence of an accident is 0.1. Therefore, the explosion rate, n, is 5.5 x 10^{-9} explosions/mi.

There are currently no routine shipments of explosives on Route 4 of a quantity that would place the ISA at risk. However, such shipments could occur during the facility's lifetime. For this analysis, the number of such shipments per year, f, was conservatively assumed to be six (H&R 522-1). The exposure distance, s, for the ISA was calculated to be 0.51 mi based on a building length of 257 ft, a setback from Route 4 of 1,155 ft, and a safe distance of 1,683 ft.

With these data, the exposure rate, r, was conservatively calculated to be 4.36 x 10^{-8} events/yr. As this is less than 10^{-6} events/yr, no additional analysis is required.

Based on the above analyses, it is concluded that explosive shipments on roadways and railways, controlled and not controlled by DOE, do not represent a risk to the ISA.

D1.7 NEARBY FACILITIES

Accidents in certain facilities in the 200 East Area (Figure D1-3) have the potential to impact the ISA facility and its operations, as discussed in Section D1.7.1. Conversely, certain nearby facilities can potentially be affected by accidents in the ISA, as discussed in Section D1.7.2.

Threats to the ISA from nearby facilities that are not known at this time will be subsequently evaluated by the unreviewed safety question process. This also applies to those activities that have been identified but that may change significantly in terms of a potential increased risk to the facility.

D1.7.1 Potential Effects from Nearby Facilities

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

- Non-reactor nuclear and non-nuclear industrial facilities within 5 mi of the ISA, including all activities conducted in and near the 200 East Area
- Nuclear reactors within a 50-mi radius of the ISA
- Military activities.

D1.7.1.1 Hazards to the Interim Storage Area from Non-Reactor Nuclear Facilities

Facilities currently operating, recently operating, or with potential to operate at the 200 East and 200 West Areas were screened along with the area between the 200 East and 200 West Areas. In this document, selected facilities were those believed to pose the most risk to safe operations at the ISA. Safety analysis reports and accident analyses prepared for these facilities were reviewed to determine possible hazards (e.g., radiological doses to personnel resulting from direct radiation, release of airborne radioactivity, or exposure to toxic chemicals).

Considered, but not included in this FSAR Annex, were the 200 East Area Burial Grounds, the Liquid Effluent Retention Facility, and the 200 Areas Effluent Treatment Facility in the 200 East Area. In the 200 West Area, the T Plant, U-Plant, Reduction-Oxidation (REDOX) Plant, and the 222-S Laboratory were considered, but not included. These facilities have insufficient radiological or toxicological inventories in a dispersible form to represent a risk to the ISA operation. The U-Plant and REDOX facilities have been shutdown for many years and are awaiting decontamination and decommissioning.

The specific facilities discussed here include the CSB, Plutonium-Uranium Extraction (PUREX) Facility, the Grout Treatment Facility, B Plant, the Waste Encapsulation and Storage Facility (WESF), the Tank Farm facilities, 242-A Evaporator/Crystallizer, Plutonium Finishing Plant (PFP), and the Low-Level Waste Disposal Site. The worst-case scenarios for each of these facilities may challenge ISA habitability. Requirements for safely performing response actions are addressed in the CSB emergency response procedures. It should be noted that the following discussions address the worst-case scenarios, and that safety-class and safety-significant SSCs and administrative controls have been provided at each facility to prevent or mitigate these potential accidents. During much of the ISA's 40-year life, the facility will not be particularly sensitive to these hazards, as it will be simply a storage facility and will require little human presence.

Canister Storage Building. The CSB is located 0.25 mi. east of the ISA facility. Hazards from the CSB are described in detail in HNF-3553, Volume 1, Chapter 3.0, and Annex A, Section A3.3. The analysis identifies hazard sources, hazardous conditions, potential accident scenarios and their initiators, and preliminary assessments of event frequencies and consequences. Hazards are identified by form and location and represent a complete spectrum of events that could occur throughout the facility and have an impact on the ISA. The CSB has been assigned a final designation of hazard category 2 facility based on material at risk.

Six potential design basis accidents were analyzed for the CSB, as follows:

- Rearrangement of Multi-Canister Overpack (MCO) internals
- Gaseous release from the MCO
- MCO internal hydrogen explosion
- MCO external hydrogen explosion
- Thermal runaway reactions inside the MCO
- Violations of design temperature criteria.

The maximum 24-hour unmitigated 100 m dose (from the CSB) resulting from these design basis accidents was determined to be 300 rem and has the potential to require the evacuation of any personnel at the ISA facility.

Plutonium-Uranium Extraction Facility. The PUREX facility is located in the 200 East Area, 1.5 mi east-southeast of the ISA. It is the most recently constructed of the irradiated fuel separation facilities and was used for processing N Reactor fuel. The principal product was a solution of plutonium nitrate that was transferred to the PFP for further processing. Another product was uranyl nitrate solution, which was processed at the Uranium Trioxide Plant. The facility has been cleaned out to the extent practical and has been transitioned to surveillance and maintenance mode awaiting a final decision on disposal of the facility. The processing canyon area and the tunnels still contain loose contamination from previous processing activities. If the status of PUREX changes (i.e., active decontamination and decommissioning), potential impacts to the ISA will be re-evaluated.

The bounding accident for the PUREX facility in the surveillance and maintenance mode is a seismic event that releases a portion of the residual contamination in the ventilation gallery above the canyon. This accident would not impair the ability of personnel at the ISA facility to perform required safety actions because the quantity of material available for release is small.

Grout Treatment Facility. The Grout Treatment Facility is located on the eastern perimeter of the 200 East Area, approximately 2.7 mi east of the ISA. The Grout Treatment Facility combined tank wastes with grout-forming solids to form a grout slurry. The waste feed stream constituent of this slurry consisted of low-level fractions of radioactive wastes. The slurry was pumped into near-surface, concrete-lined vaults for permanent disposal. Only one vault has been filled with grout (completed in 1987) even though additional vaults have subsequently been designed and constructed. This facility is currently not operational but has the potential to operate again in the future.

If the Grout Treatment Facility resumes operation, the maximum credible accident release postulated involves a double-ended jumper leak on the grout feed line, with the pit cover blocks left off. This postulated release would spray low-level waste as an aerosol for a maximum period of 24 hours before detection by a visual inspection. If the status of the Grout Treatment Facility changes (i.e., begins operations), potential impacts to the ISA will be re-evaluated. If the ISA habitability is challenged by an event at the Grout Treatment Facility, the appropriate action (i.e., notification, take-cover, emergency shutdown, evacuation) will be initiated using CSB emergency response procedures.

B Plant Facility. The B Plant is located approximately 1,500 ft east of the ISA. Until 1952, B Plant was operated as a fuel separation facility. In 1968, it was converted to a waste fractionation plant to remove ^{137}Cs and ^{90}Sr from radioactive waste streams. This had the effect of reducing heat loads in the double-shell tanks. The B Plant currently is deactivated and has been transitioned to surveillance and maintenance mode.

The worst-case credible accident at B Plant is a postulated flooding of the 291-B high-efficiency particulate air filters and a subsequent hydrogen explosion. According to HNF-SD-WM-BIO-003, *B Plant Basis for Interim Operations*, the dose to the collocated worker at 100 m from B Plant was calculated to be 952 rem, and the maximum offsite dose to be 0.368 rem at the Columbia River, 16.8 mi east of B Plant. Systems and administrative controls are in place at B Plant to mitigate the releases. The worst-case credible accident at the B Plant involves the filters, until they are decontaminated and decommissioned. This scenario has the potential to require evacuation of any personnel at the ISA facility. CSB emergency response procedures will describe the action to be taken by personnel at the ISA facility in the event take-cover or evacuation actions are required by events at nearby facilities.

Waste Encapsulation and Storage Facility. The WESF is distinct from B Plant, even though it is located on the west end of B Plant and shares a common wall with the plant. Historically, WESF was involved in converting the cesium and strontium removed from waste streams at B Plant into cesium chloride and strontium fluoride salts. These materials were then encapsulated in double-walled metal containers and stored in a water-filled cooling basin. Strontium fluoride and cesium chloride capsules are still being stored in this fashion at WESF, but no new capsules are being produced.

According to HNF-SD-WM-BIO-002, *Waste Encapsulation and Storage Facility Basis for Interim Operation*, the worst-case credible accident postulated at WESF involves a loss of cooling water in the storage pool, with a subsequent rupture of capsules and aerosolization of the capsule contents. The maximum dose to the public receptor at the near bank of the Columbia River, about 18 miles from WESF, is calculated to be 9 rem. The 24-hour unmitigated dose at 100 m is 1,700 rem from inhalation and 32 rem from radiation shine. Systems and administrative controls are in place at WESF to mitigate the releases. However, this scenario has the potential to require evacuation of ISA operating personnel. CSB emergency response procedures will describe the action to be taken by personnel at the ISA facility in the event take-cover or evacuation actions are required by events at nearby facilities.

Tank Farm Facilities. The Tank Farm facilities include 149 single-shell tanks and 28 double-shell tanks for the storage of liquid radioactive mixed waste solutions from the Hanford Site's chemical processing plants and associated support facilities, which include holding tanks, transfer lines, and valve pits. The Tank Farm facilities are located in both the 200 East and 200 West Areas.

The Tank Farm facilities nearest to the ISA are the B, BX, and BY Tank Farms, located approximately 0.8 mi northeast of the ISA. Each of the farms has 12 single-shell tanks. The B and BX single-shell tanks are rated at 500,000 gal each, and those in the BY Tank Farm are rated at 750,000 gal each. In addition, there are four 55,000-gal tanks in the B Tank Farm, and a double-contained receiver tank serves the BX and BY Tank Farms. The tanks in these tank farms were constructed between 1943 and 1949.

Tank Farm operations are evaluated in HNF-SD-WM-SAR-067, *Tank Farms Final Safety Analysis Report*. The accident analysis results, together with the results of the hazard analysis, form the basis for control decisions that define a suite of safety SSCs and technical safety requirement controls to eliminate or reduce the risk of identified potential hazardous conditions and postulated accidents for Tank Farm facilities and operations. With the exception of three accidents, the selected safety SSCs and technical safety requirement controls are shown by the accident analysis results to manage the risk below the Tank Farm risk guidelines.

Additional Tank Farm design and operational features identified as defense-in-depth controls further enhance safety. For postulated flammable gas deflagrations, organic solvent fires, and seismic events, the conservative accident analysis results exceed the Tank Farm risk guidelines even with the identified controls. The risks from abnormal events and postulated accidents result from the uncontrolled release of radioactive or hazardous materials stored in the tanks.

Waste transfers into Hanford Site single-shell tanks are no longer allowed. Most of the tanks in these three farms have been interim stabilized, meaning that as much of the liquid has been removed from the waste as practicable, leaving solids with a minimum liquid content. Administrative controls are in place at the tank farms for early detection and mitigation of a release caused by mis-transfer during waste transfer operations. Because of the site distance (0.8 mi) and the controls in place, these scenarios would not significantly impact the ISA. Even though the cross-site transfer line passes near the ISA, a leak from the line is not considered a credible event because of its safety-class, encased-pipe design (HNF-SD-WM-SAR-067).

If there was a release from a Tank Farm accident, the immediate action for personnel at the ISA facility would be to respond to the take-cover sirens. Depending on the specific conditions, follow-up actions could include evacuation per CSB emergency response procedures.

Eventual retrieval of the tank wastes for final disposal is planned for the future. Activities involving waste removal and decontamination and decommissioning of the tanks will be evaluated at that time for their effect on the ISA safety analysis.

242-A Evaporator/Crystallizer. The 242-A Evaporator/Crystallizer is located 1.5 mi east of the ISA. The 242-A Evaporator/Crystallizer uses evaporative concentration to reduce the volume of liquid wastes. The concentrated slurry, reduced in volume, is transferred and stored in underground double-shell waste storage tanks. The process condensate is routed to the Liquid Effluent Retention Facility for storage and treatment at the Effluent Treatment Facility.

The worst-case accident scenario reported in HNF-SD-WM-SAR-023, *242-A Evaporator Safety Analysis Report*, involved a release from a spray leak in the pump room and a failure of the exhaust system's high-efficiency particulate air filters. Although this event is considered extremely unlikely, the event was analyzed for safety-class determinations at the 242-A Evaporator/Crystallizer. The release from a spray leak is comprised of a liquid component and an aerosol component. The analysis assumed that the liquid component would remain in the pump room; however, the aerosol component would be released to the environment via the K1 exhaust stack. This scenario might require action to prevent exposure of ISA operators to the release (i.e., take cover), but it is not likely to require shutdown of ISA operations. CSB emergency response procedures will describe the action to be taken by personnel at the ISA facility in the event take-cover or evacuation actions are required by events at nearby facilities.

Plutonium Finishing Plant. Located near the western boundary of the Hanford Site in the 200 West Area, the PFP is 5.0 mi west of the ISA. The PFP converts plutonium nitrate solution to plutonium oxide and performs plutonium handling and storage operations. Contaminated liquid waste streams from the PFP are routed to the tank farms. This facility is in transition from its previous special nuclear material processing mode to preparation for decontamination and decommissioning.

During its previous mission of processing special nuclear materials, plutonium-bearing materials were produced and still exist in the facility. These materials include pure and mixed oxides, fluorides, oxalates, silicates, and organic-based sludge and residues. In HNF-SD-CP-SAR-021, *Plutonium Finishing Plant Final Safety Analysis Report*, the current PFP missions are defined as (1) receiving, sorting, storing, and shipping special nuclear material, (2) stabilizing the reactive materials that remain in the plant, (3) providing laboratory support for other Hanford Site facilities, (4) handling radioactive and mixed waste, and (5) shutdown facility surveillance.

The bounding accident for the PFP is a seismic event with a horizontal ground acceleration of 0.2 *g*. The consequences for this event are calculated to be 15.2 rem effective dose equivalent at the nearest occupied facility in the worst-case direction, 1,800 ft west-northwest of PFP, and 0.31 rem at the Site boundary, 7.8 mi west. Because of the distance separating the ISA from the PFP, the release would not adversely affect ISA operations and would not require evacuation of personnel at the ISA facility.

Low-Level Waste Disposal Site. The commercial low-level waste disposal site operated by U.S. Ecology, Inc., is the only non-DOE industrial facility within 5 mi of the ISA (Figure D1-1). The disposal site is on land leased from Washington State located 1 mi southwest of the ISA. The low-level waste is buried in U.S. Department of Transportation-approved shipping containers. Monitoring of groundwater and vegetation is performed as required by the facility's NRC operating license and environmental impact statement. This facility is not likely to have a significant accidental airborne radioactive release that could adversely affect the ISA.

D1.7.1.2 Hazards to the Interim Storage Area from Non-Nuclear Hanford Site Facilities

A number of non-nuclear industrial facilities operating in the 200 Areas pose the potential for accidental fires, explosions, or releases of toxic fumes. These include the Essential Materials Warehouse (Building 275-EA), oil and paint storage buildings, fabrication shops, gas cylinder storage buildings, the spare parts and electrical warehouse, B Plant storage buildings, maintenance facilities, gasoline service stations, and the powerhouse complexes in each area (284-E and 284-W). Considering its location, Building 275-EA may have the potential to pose a risk to ISA operations and personnel and is detailed below. The 2,000-lb chlorine bottles previously located at Building 283-E have been removed and are no longer a hazard to the ISA.

Building 275-EA. Building 275-EA, the Essential Materials Warehouse, was constructed in 1955 and is located near the PUREX facility about 1.5 mi east-southeast of the ISA. It is classified as an unprotected wood frame structure and is susceptible to collapse as a result of an external event (e.g., earthquake, wind, snow, or ash loading) or an internal event (e.g., forklift collision with a bearing wall, or fire). Building 275-EA currently stores more than 100 different types of potentially hazardous solids and liquids including acids, bases, solvents, fluorides, pesticides, and herbicides. Radioactive materials are not stored in this building.

The worst-case chemical release postulated in the safety analysis occurs following a building collapse whereupon 2,450 kg of 1,1,1-trichloroethane evaporates under adverse atmospheric conditions. The maximum concentration at the PUREX facility gatehouse and environs is calculated to be 187 ppm, which is below the time-weighted average threshold limit value of 350 ppm (i.e., the concentration for this chemical that an industrial worker may be repeatedly exposed to without adverse effects), as given in *Threshold Limit Values and Biological Exposure Indices for 1989-1990* (ACGIH 1989). Concentration levels at the ISA would be significantly less than this because of the 1.5 mi separation.

D1.7.1.3 Hazards to the Interim Storage Area from Nuclear Reactors

Three recently operating reactors, the N Reactor, Fast Flux Test Facility (FFTF), and the Critical Mass Laboratories, no longer pose a threat to the ISA. The N Reactor is undergoing decontamination and decommissioning, the FFTF is in standby mode and may operate again in the future, and the Critical Mass Laboratories in the 200 East Area north of the PUREX facility are currently being used as tank farm office areas.

The N Reactor was a 4,000 MW, dual-purpose, pressure tube, light-water cooled, graphite-moderated reactor (UNI-M-90, *N-Reactor Updated Safety Analysis Report*). It is located in the 100 N Area and is about 5.5 mi from the nearest Hanford Site boundary and about 9 mi. from the ISA. The N Reactor began operating in 1964 and produced plutonium for the defense program and steam for electrical power generation. The reactor was shut down in 1987 for safety improvements and then subsequently defueled and placed in cold standby in 1988. It is currently being decontaminated and decommissioned. Current plans are to remove the reactor building structures for the production reactors down to the reactor block and then cocoon the reactor block for 75-year safe storage.

The FFTF is a 400 MW, sodium-cooled, mixed-oxide-fueled reactor that is currently in standby status with the fuel removed from the core. Sodium is kept circulating in the loops at 400 °F until a decision is made to either restart the reactor for future missions or decommission it. The FFTF is located in the 400 Area and is approximately 4.5 mi from the nearest Hanford Site boundary, which is to the east of the facility (Figure D1-2) and about 13.5 mi. from the ISA. Spent FFTF fuel, removed from storage in liquid sodium and packaged in double-walled casks with an inert cover gas, is stored in the ISA at the 400 Area.

According to HEDL-TI-7500-FSAR, *Fast Flux Test Facility (FFTF) Final Safety Analysis Report*, the bounding accidents for the FFTF in its current status involve (1) a liquid sodium spill and (2) damage to a cask and its contents in the ISA. The sodium spill scenario postulates a spill of 180,000 lb of activated liquid sodium. The 2-hour dose at 1.5 mi from the FFTF is 0.015 mrem. The dose at 4.5 mi from the facility for a 30-day exposure is 0.26 mrem. The maximum credible cask release at the 400 Area ISA postulated cracking in 100% of the fuel pins of one cask, with crushing and exposure of 1% of the fuel material. The dose at 100 m from the facility was 4.5 rem, the maximum dose at the Site boundary was 4.0 mrem. Because of the distance from the 200 Area ISA, these accidents do not pose a hazard to ISA operations.

The only operating nuclear reactor on the Hanford Site is the Columbia Generating Station (formerly known as Washington Nuclear Plant 2). The location of this reactor is shown in Figure D1-2. The Columbia Generating Station is an operating commercial nuclear power plant using a boiling-water reactor steam supply system. The design power level was increased to 3,486 MW in 1995, as noted in Docket No. 50-397, *Final Safety Analysis Report, Washington Nuclear Power Plant No. 2*. The reactor was designed by the General Electric Company and is designated as a BWR/5 with a Mark II containment.

By the requirements of Title 10, *Code of Federal Regulations*, Part 100, "Reactor Site Criteria" (10 CFR 100), the following are the maximum allowable doses for the Columbia Generating Station:

Location	Duration	Whole body dose	Thyroid
Exclusion area boundary	2 hours	250 mSv (25 rem)	3,000 mSv (300 rem)
Low population zone	30 days	250 mSv (25 rem)	3,000 mSv (300 rem)

The exclusion area boundary for the Columbia Generating Station is 1.2 mi, and the low population zone distance is 3 mi. The ISA is located approximately 11 mi from the Columbia Generating Station. Using the atmospheric diffusion guidance provided in NRC Regulatory Guide 1.3, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors*, to estimate the dose reduction as a function of distance, it was determined that the 2-hour and 30-day doses at the ISA would be reduced by a factor of 20. Review of the site-specific meteorology provided in the FSAR for the Columbia Generating Station (Docket No. 50-397) shows there would be no significant reduction for the change in wind direction. The factor of 20 reduction for distance would result in a whole body dose of 12.5 mSv (1.25 rem) and a thyroid dose of 150 mSv (15 rem). The expected dose that would be received at the ISA if a loss of coolant accident occurs would be significantly less than this. NRC Regulatory Guide 1.3 requires an assumption that 25% of the radioactive iodine and all of the noble gases are released to the containment. In fact, the emergency core cooling system would prevent most of these releases, as little fuel damage would occur as a result of the loss of coolant accident.

Energy Northwest (formerly known as Washington Public Power Supply System) plans to add an independent spent fuel storage installation on their leased property. The independent spent fuel storage installation will be licensed following the requirements in 10 CFR 72. According to 10 CFR 72.106, an individual located at the installation's controlled area boundary shall not receive a dose greater than 5 rem. At the ISA, this would result in a dose not exceeding 0.25 rem. CSB emergency response procedures will describe the action to be taken by personnel at the ISA facility in the event take-cover or evacuation actions are required by events at nearby facilities.

D1.7.1.4 Hazards to the Interim Storage Area from Industrial Facilities Off the Hanford Site

There are no oil or gas pipelines in the vicinity of the ISA. The nearest major natural gas pipeline to the ISA site is about 29 mi. A 20-in. gas transmission line of the Northwest Pipeline Corporation is located east and essentially parallel to U.S. Highway 395 between Pasco and Ritzville, Washington. A second pipeline system consisting of parallel 36-in. and 42-in. lines, owned by Pacific Gas Transmission Company, passes through Wallula, approximately 33 mi from the site (Hosler 1996). These distances eliminate any potential hazard to plant operations from a natural gas fire or explosion.

The nearest petroleum product storage tanks are located 38 mi from the site. These tanks include 23-million-gallon capacity tanks at the Chevron Pipeline Company and 21-million-gallon capacity tanks at the Tidewater Barge Lines. There are no plans to use a third petroleum storage facility at the Port of Pasco (Hosler 1996).

Located within the Richland city limits is the Framatome ANP (formerly known as Siemens Power Corporation) Richland Engineering and Manufacturing Facility. All operational steps for the manufacture of nuclear fuel for light water reactors are conducted within the facility, including the conversion of UF_6 to UO_2. The most limiting postulated accident at this facility is a fire in the UF_6 cylinder storage area (Siemens 1994, *Siemens Power Corporation Richland Engineering and Manufacturing Facility Emergency Plan*). Fusible plugs in twelve cylinders are assumed to melt causing the release of UF_6, which reacts with moisture in the air to form UO_2F_2 (solids in the form of uranyl fluoride hydrates) and 4HF (as hydrogen fluoride gas). UF_6 is a radiological hazard by inhalation. However, UF_6 is also a concern because of its chemical toxicity and the associated HF that can cause skin and eye burns and lung impairment. This accident results in a dose that exceeds 10 mSv (1 rem) out to 1.2 mi. The accident also exceeds the Emergency Response Planning Guide-2 toxicology limits out to 8.8 mi (ACGIH 1989). The Emergency Response Planning Guide-2 is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action. The ISA is located 19 mi from the Framatome ANP facility. Therefore, operators would not be placed at risk by an accident at the Framatome ANP facility.

No other non-nuclear industrial facilities or operations have been identified that can impact ISA operations.

D1.7.1.5 Hazards to the Interim Storage Area from Military Facilities

The Yakima Firing Center is a sub-installation under the command of Fort Lewis (Tacoma, Washington). Further information is given in the *Final Environmental Impact Statement — Ft. Lewis Military Installation* (DOA 1979). The southeastern boundary of the Yakima Firing Center is located about 23 km (14 mi) from the ISA (see Figure D1-12). The Yakima Firing Center is used for military maneuvers and weapons training and is the only significant military activity in the vicinity of the Hanford Site.

The only weapon currently in use at the Yakima Firing Center known to present a hazard to the Hanford Site is the Multiple Launch Rocket System (MLRS). With a range of approximately 16 mi, the MLRS could potentially impact the ISA site. However, the MLRS is only fired from the perimeter of the Yakima Firing Center into a centrally located impact zone. The safety fan for the MLRS is shown in Figure D1-12. The MLRS is fired away from the Hanford Site and is only fired with dummy warheads. Given this information, additional safety features, and the administrative controls in place at the Yakima Firing Center, a weapons accident having an impact on the Hanford Site is very improbable.

A more probable hazard to Hanford Site facilities is a scenario in which a fire started within the Yakima Firing Center boundary spreads to the Hanford Site. Exploding artillery shells, sparks from tracked vehicles or other machines, and careless smoking by troops might start brush fires that, under adverse meteorological conditions, could spread rapidly beyond the Yakima Firing Center boundaries. The hazards associated with range fires are discussed in Chapter 3.0 of HNF-3553, Volume 1.

D1.7.2 Potential Effects to Nearby Facilities

ISA accidents with the potential to affect the maximum onsite individual (which may include persons at some of the facilities discussed above) are discussed in Section D3.4.2. Section 15.4.2.2 of HNF-3553, Volume 1 requires that a hazards assessment for each SNF Project facility be prepared based on the facility-specific hazards and safety analyses that are contained in each SNF Project Annex. The scope of the hazards assessment is provided in Section 15.4.2.2 of HNF-3553, Volume 1. The hazards assessment characterizes the potential consequences on workers, the public, and the environment for each postulated accident and determines the emergency planning zone for each facility as well as the emergency class, protective actions, and the observable indications and criteria (emergency action levels) corresponding to the range of identified accidents. The hazards assessment provides the framework for response to virtually any declared emergency.

As stated in Section 15.4.3 of HNF-3553, Volume 1, prompt and accurate emergency notifications would be made to mitigate consequences and to protect the health and safety of workers, the public, and the environment in accordance with DOE/RL-94-02, *Hanford Emergency Management Plan*. The Emergency Operations Center is responsible for notification of affected areas and other contractors onsite. The DOE Richland Operations Office Emergency Operations Center is responsible for follow-up notifications when emergencies are reclassified or terminated in accordance with Section 15.4.3.4 of HNF-3553, Volume 1.

D1.8 VALIDITY OF EXISTING ENVIRONMENTAL ANALYSES

No significant discrepancies have been identified between the site characteristic assumptions made in this chapter and those made in the site-wide SNF Project Environmental Assessment (DOE/EA-1185).

D1.9 REFERENCES

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Figure D1-1. Location of the 200 Area Interim Storage Area Relative to the Canister Storage Building.

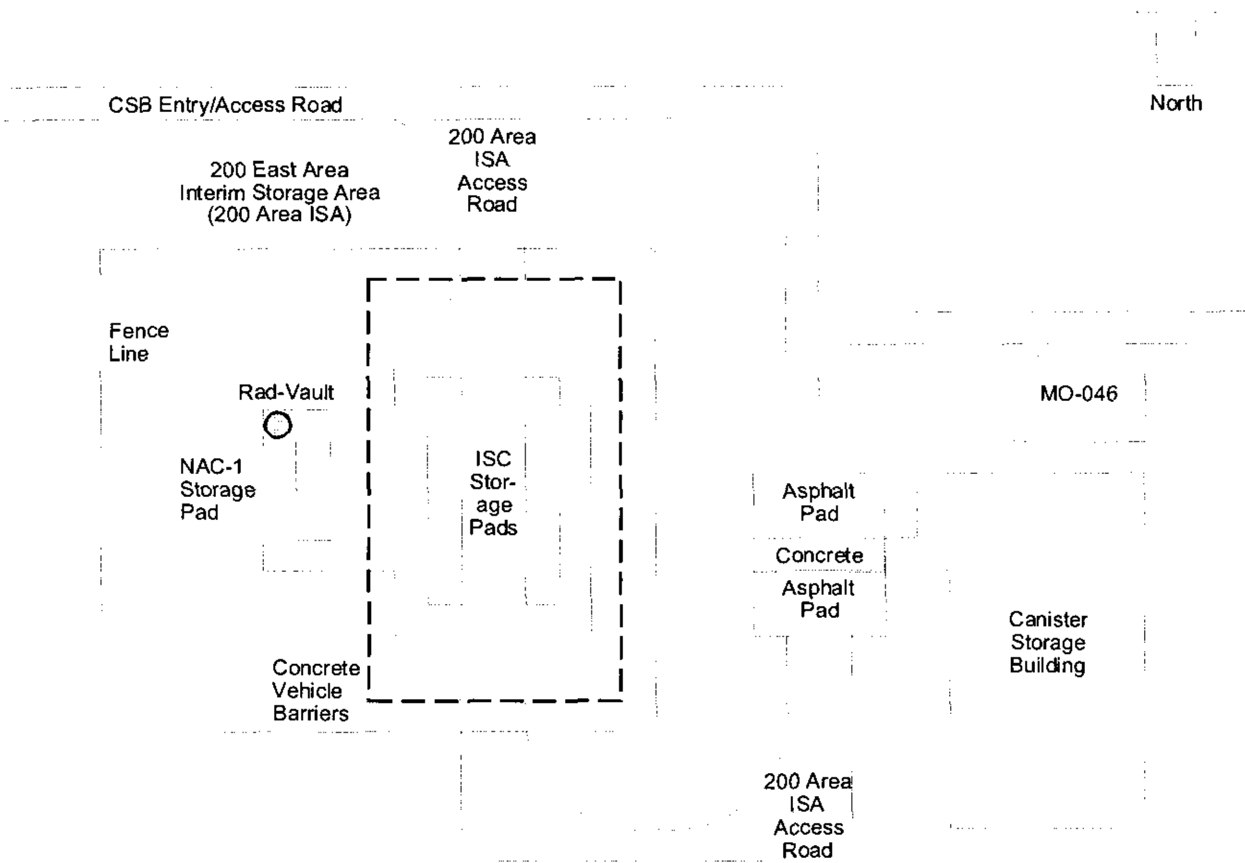
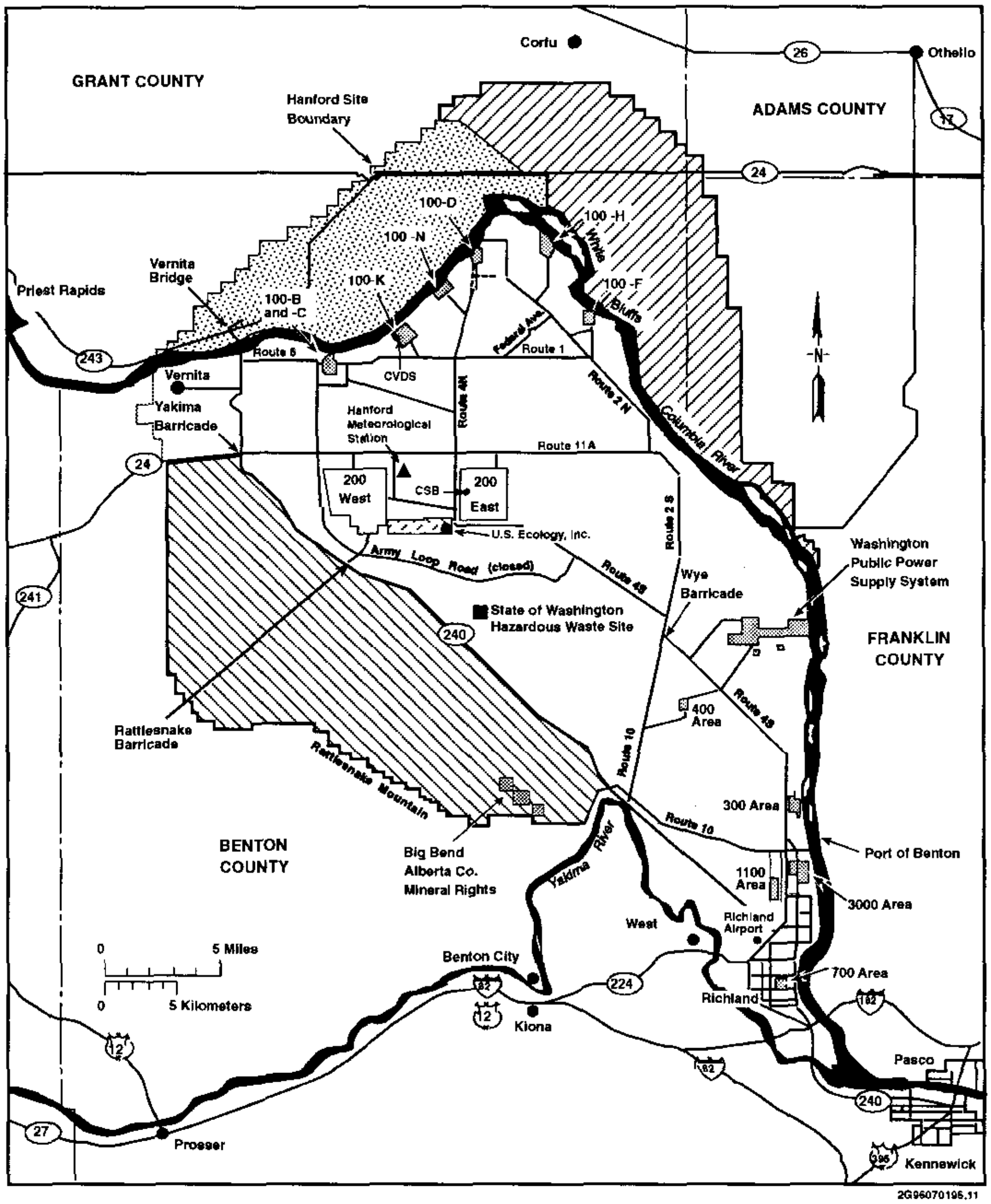


Figure D1-2. Hanford Site Boundaries.



Wahluke Wildlife Recreation Area, State of Washington Department of Game

Saddle Mountain National Wildlife Refuge, U.S. Fish and Wildlife Service

Fitzner-Eberhardt Arid Lands Ecology Reserve

State of Washington Leased Land

Figure D1-3. Onsite Population Distribution in the 200 East Area by Zone.

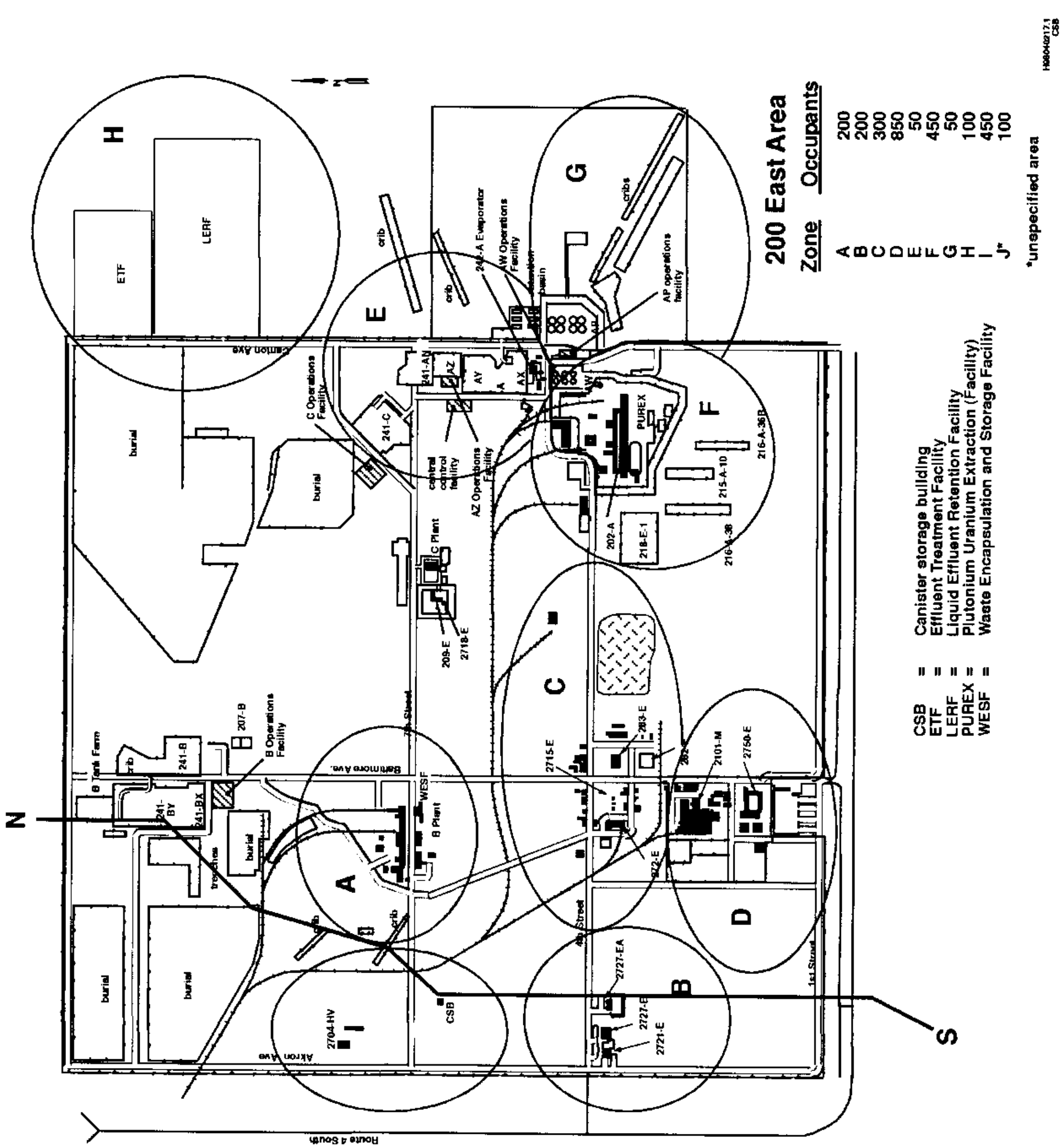


Figure D1-4. Wind Rose for the 200 East Area.

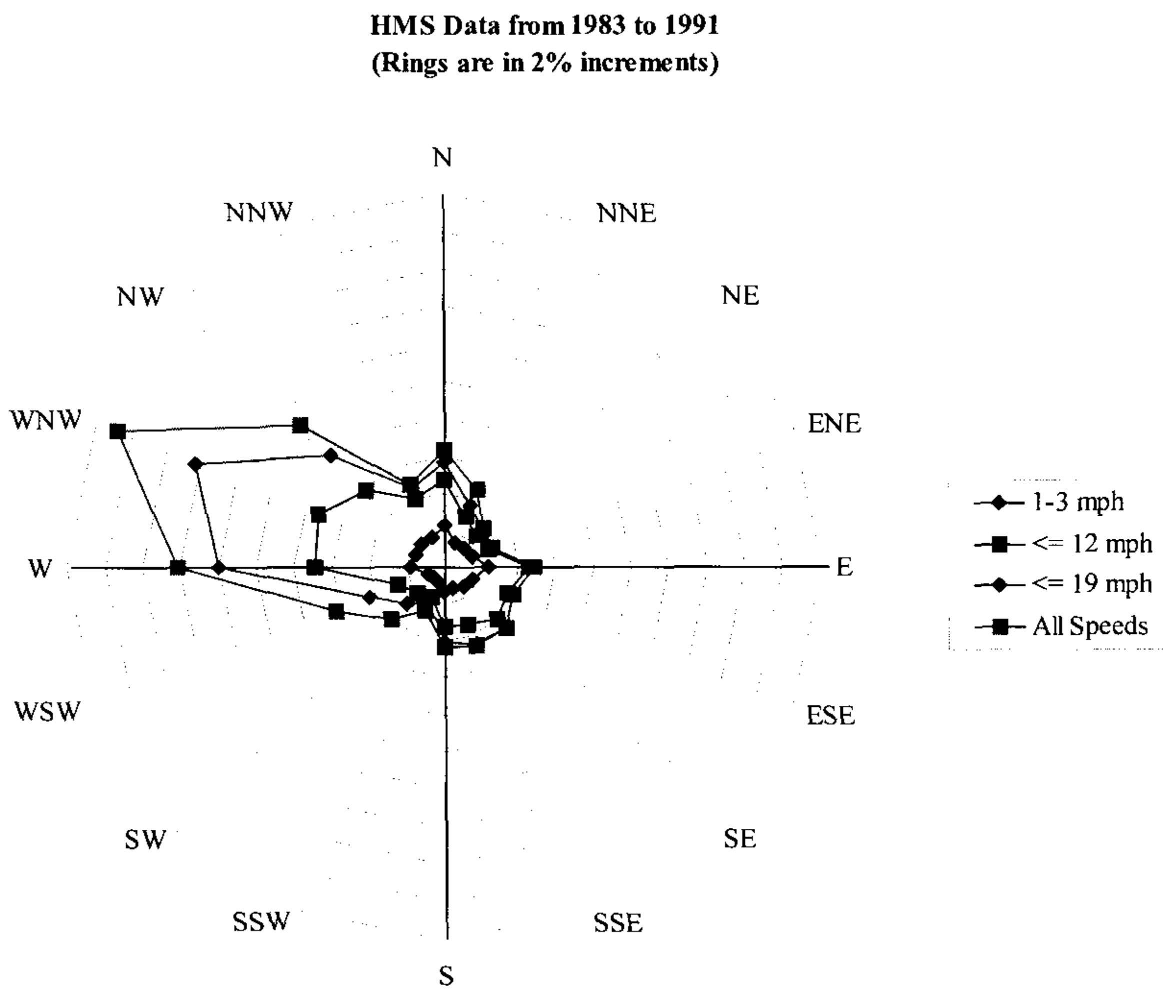


Figure D1-5. Wind Speed Histogram for the 200 East Area.

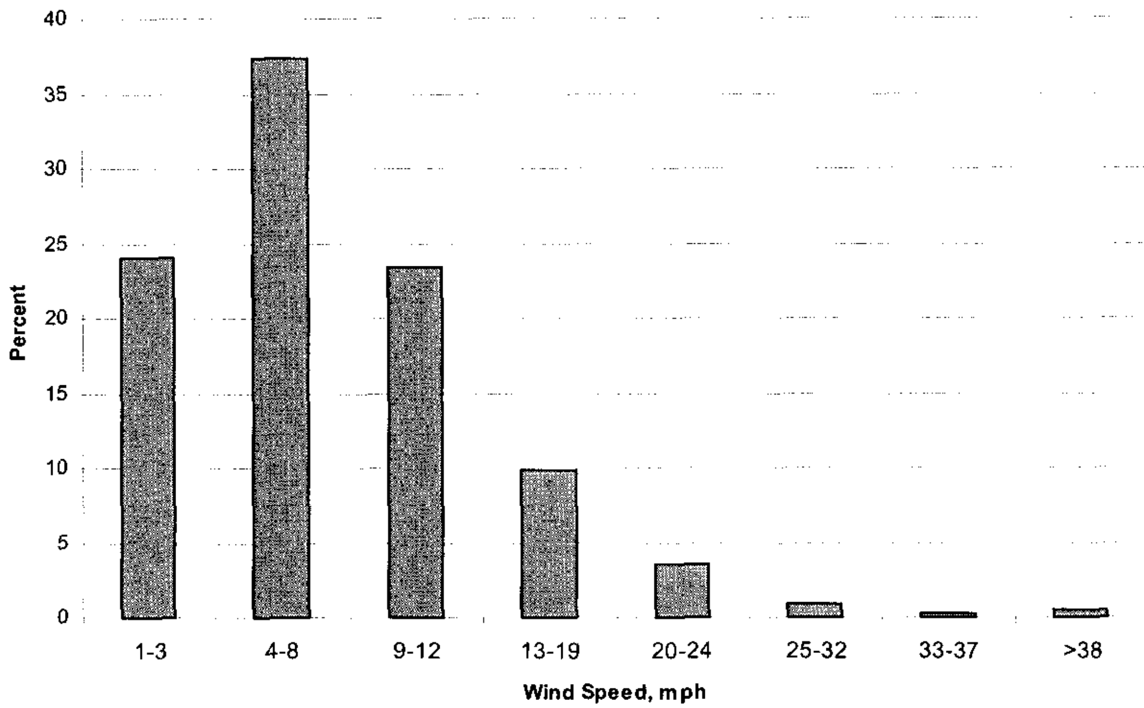


Figure D1-6. Hanford Site Topographic Map and Cross Section.

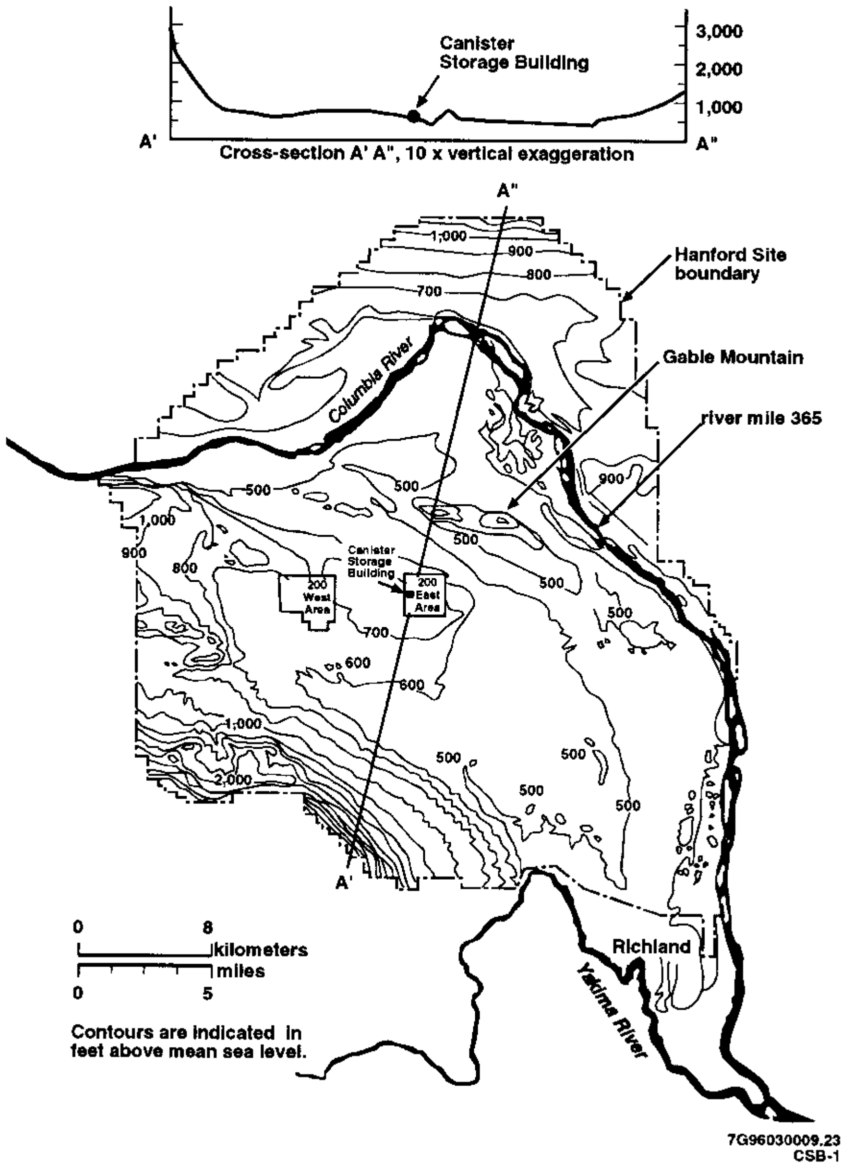


Figure D1-7. Geology of the 200 East and West Areas.

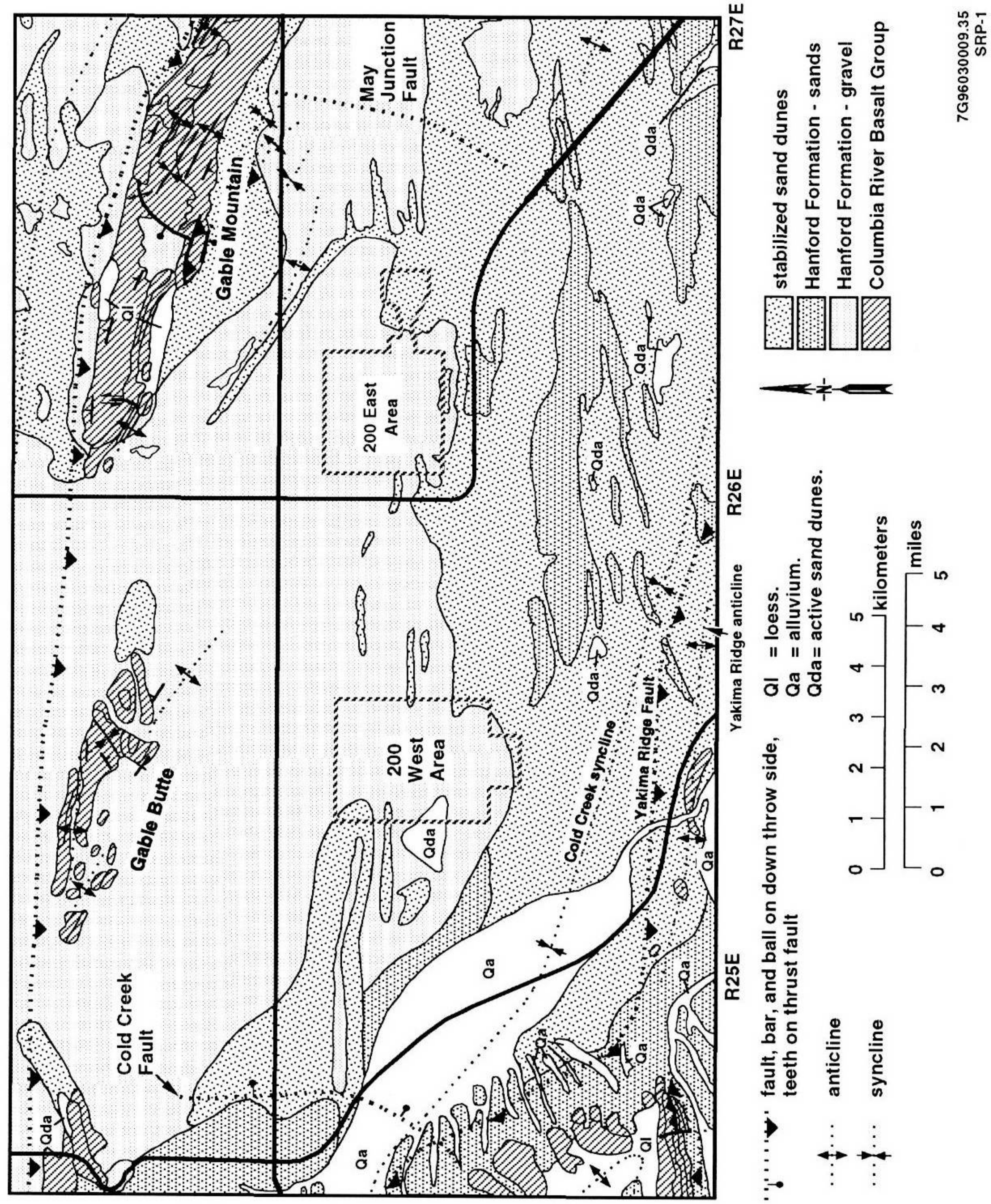


Figure D1-8. Stratigraphy of the 200 East Area.

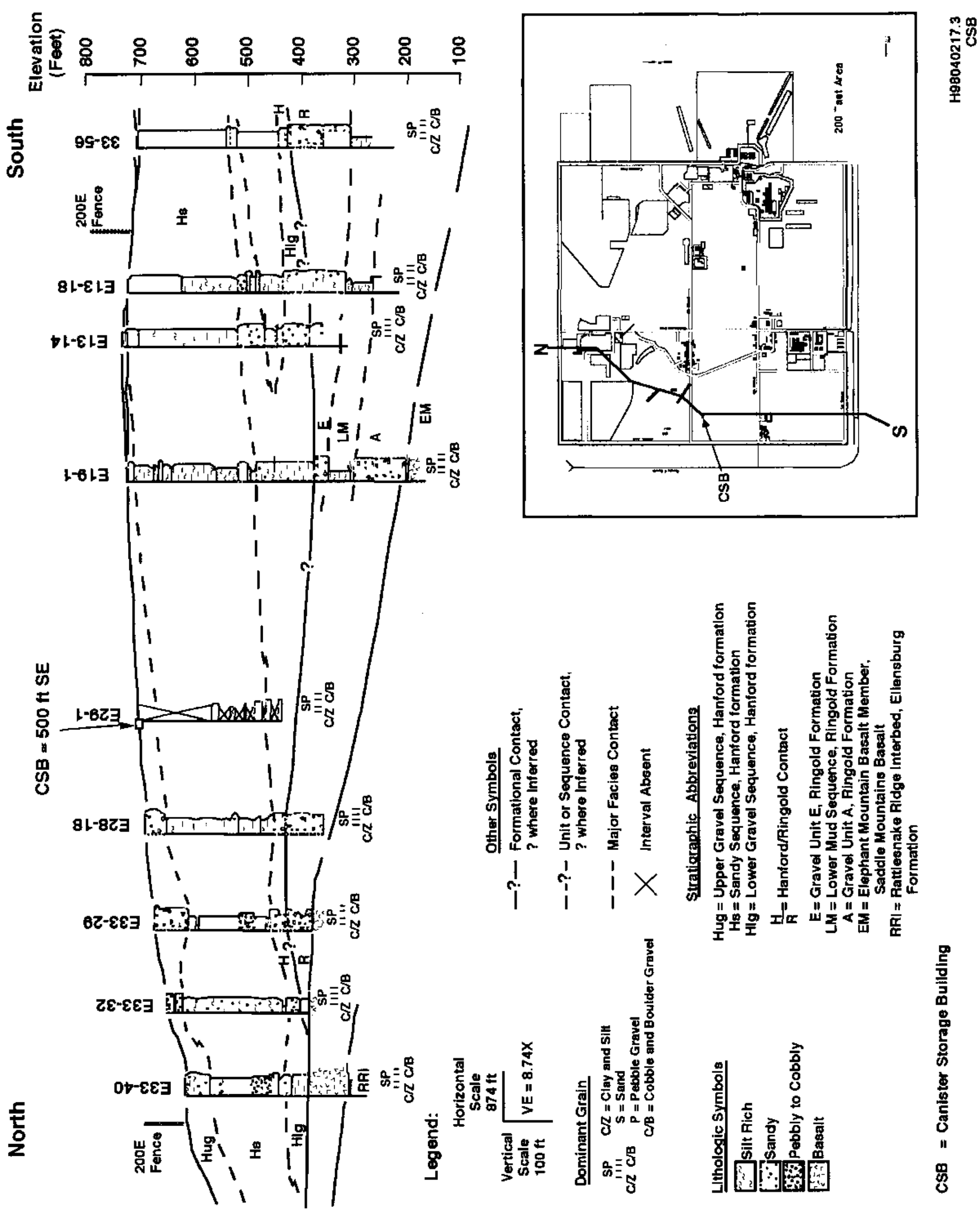
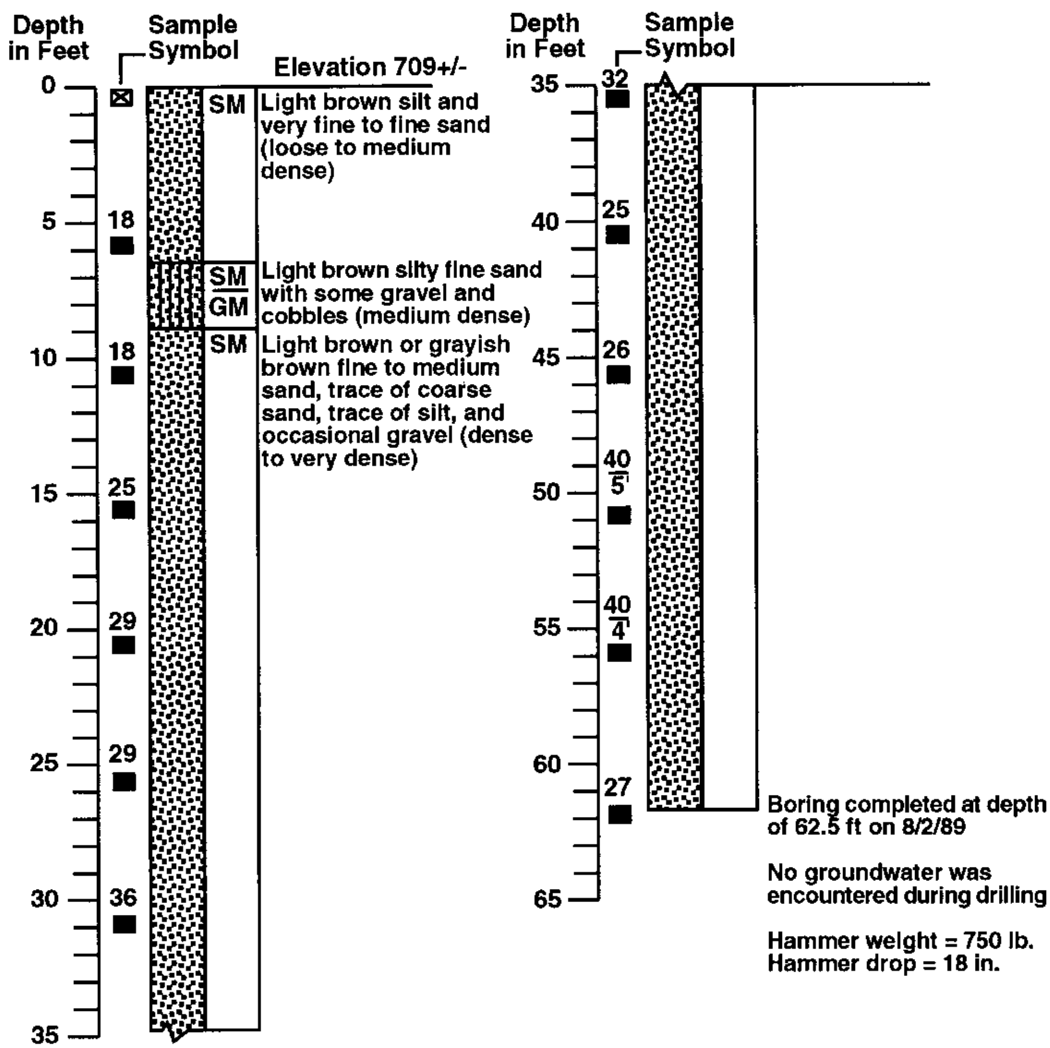


Figure D1-9. Log of Borings: Boring VP-15.



Key to Log Borings

32 Blows required to sampler one foot

■ Indicates depth at which undisturbed sample was extracted

⊠ Indicates depth at which disturbed sample was extracted

Figure D1-10. Geologic Cross-Section of the Canister Storage Building Site.

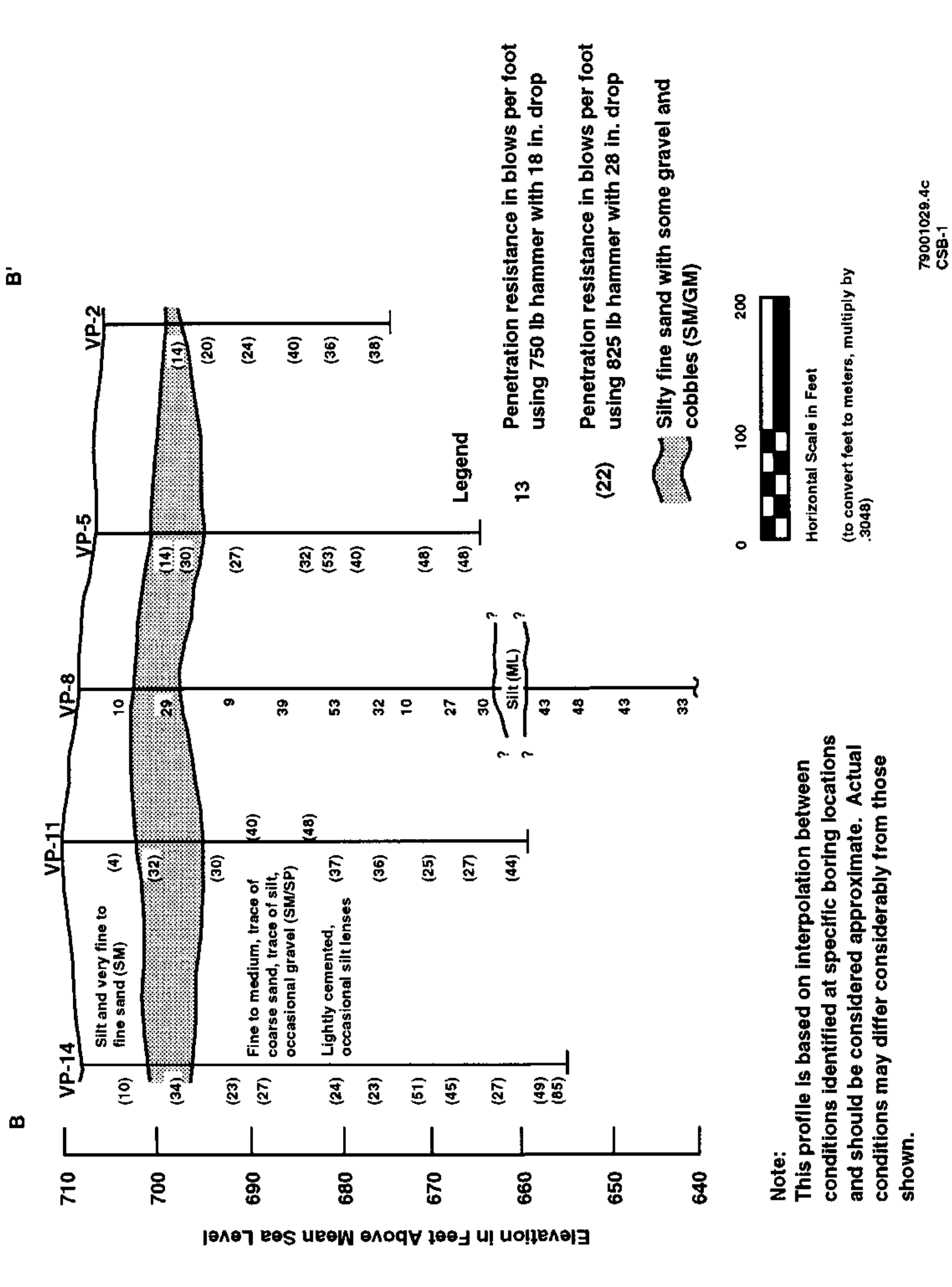


Figure D1-11. Dynamic Soil Properties of the Canister Storage Building Site.

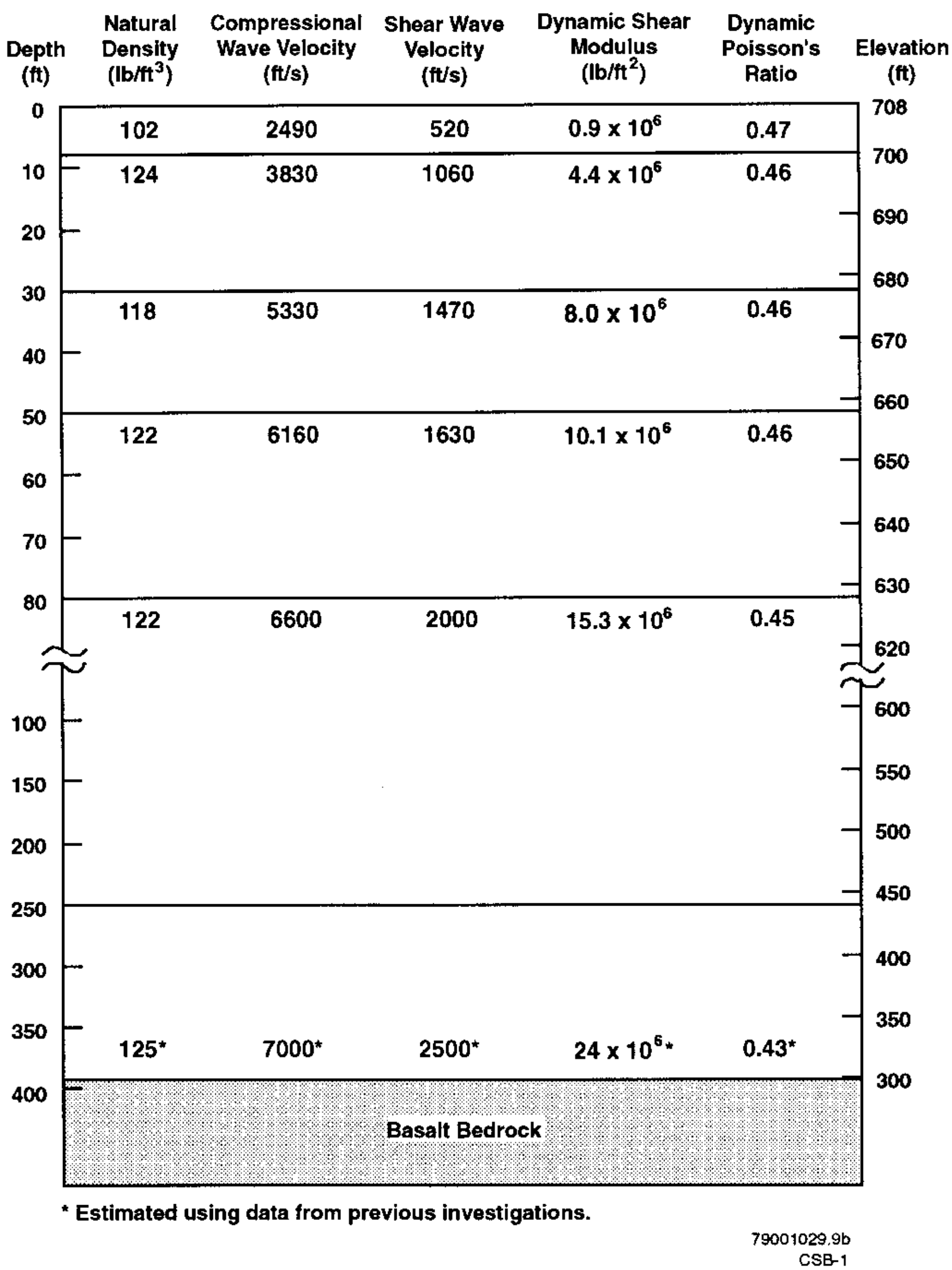
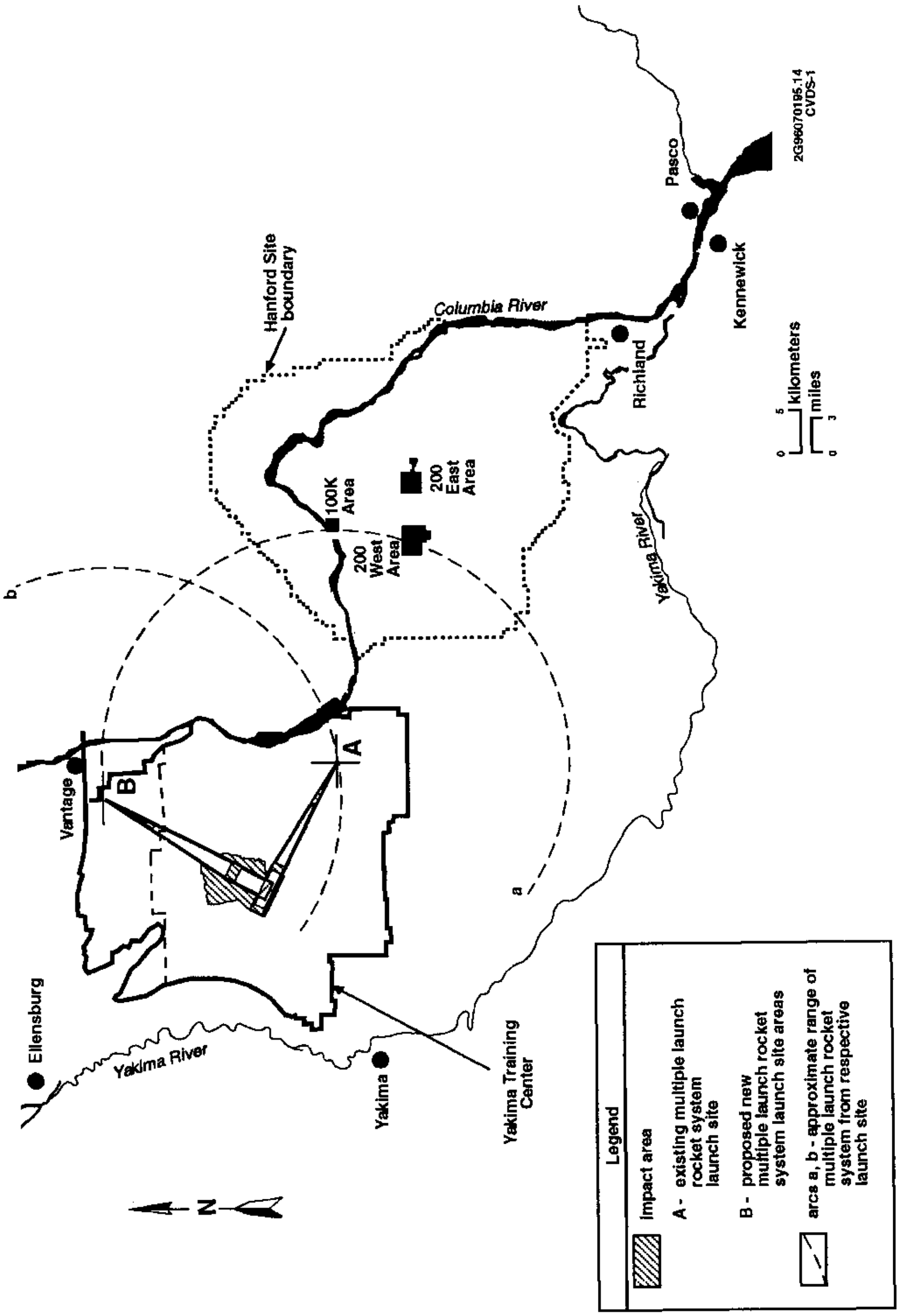


Figure D1-12. The Location of Yakima Firing Center with Respect to the Hanford Site.



CHAPTER D2.0

FACILITY DESCRIPTION

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ACI	American Concrete Institute
AFFRI	Armed Forces Fuel Research Institute
AL	aluminum
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATM	approved testing material
AWG	American wire gauge
BWR	boiling water reactor
CCC	core component container
CE	Combustion Engineering, Incorporated
CFR	Code of Federal Regulations
CSB	Canister Storage Building
DFA	driver fuel assemblies
DOE	U.S. Department of Energy
DORF	Diamond Ordnance Reactor Facility
FFCR	fuel follower control rod
FFTF	Fast Flux Test Facility
FHA	fire hazards analysis
GA	General Atomics Corporation
GE	General Electric Company
GRC	galvanized rigid conduit
ID	inner dimension
IEM	interim examination and maintenance
ISA	interim storage area
ISC	interim storage cask
ISO	International Standards Organization
LWR	light water reactor
MT	metric ton
MTHM	metric ton of heavy metal
MWd	megawatt day
NEMA	National Electrical Manufacturers Association
NRC	U.S. Nuclear Regulatory Commission
NRF	Neutron Radiography Facility
NTS	Nevada Test Site
OD	outer dimension
PNNL	Pacific Northwest National Laboratory
PTFE	polytetrafluoroethelene
PVC	polyvinyl chloride
PWR	pressurized water reactor
SNF	spent nuclear fuel
SRP	segmented rod program

LIST OF TERMS (continued)

SS	stainless steel
SSC	structure, system, and component
UBC	Uniform Building Code
WE	Westinghouse Electric Corporation
ZPA	zero-period acceleration

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D2.0 FACILITY DESCRIPTION

The 200 East Area Interim Storage Area (200 Area ISA) at the Hanford Site provides for the interim storage of non-defense reactor spent nuclear fuel (SNF) housed in dry cask storage systems. The 200 Area ISA is a relatively simple facility consisting of a boundary fence with gates, lighting along the perimeter, and pads on which to place dry storage casks. The pads are concrete and level gravel placed by the project based on HNF-2524, *200 East Area Interim Storage Area Preliminary Safety Evaluation*, approved Site Evaluation Form (#2E-96-10), and an "Approval to Construct" authorization documented in Letter 98-SFD-100/9853923, *Contract No. DE-AC06-96RL13200 - 200 Area Interim Storage Area (ISA) Construction Authorization* (Hansen 1998). The fence supports safeguards and security and provides a radiation protection buffer zone.

An equipment storage building (pre-engineered metal building) is proposed near the 200 Area ISA to house lifting devices, impact limiters, transfer casks, and various other types of equipment associated with storage, movement, or transport of the casks. This storage building will be located outside the 200 Area ISA and provides no safety-related function.

The three different dry cask storage systems used at the 200 Area ISA are as follows:

- Interim Storage Cask (ISC) used for the Fast Flux Test Facility (FFTF) SNF.
- Neutron Radiography Facility (NRF) TRIGA[1] casks and DOT-6M[2] containers within a Chem-Nuclear Services, Incorporated, Rad-Vault[3] storage vault used for NRF TRIGA SNF.
- NAC-1 casks[4] within International Standards Organization (ISO) containers used for commercial light water reactor (LWR) SNF from the 300 Area.

The 200 Area ISA is located approximately 0.25 mi west of the Canister Storage Building (CSB). The footprint of the ISA is nominally 500 ft by 400 ft surrounded by a chain-link fence, with gates in the fence that control access of vehicles and personnel. Light poles provide illumination for the 200 Area ISA. Within the fenced area are concrete pads for placement of the ISCs and the NAC-1 casks within the ISO containers. The Rad-Vault holding NRF TRIGA casks and DOT-6M containers is to be placed on a level gravel pad placed by the project.

Interim storage at the ISA is intended until shipment of the materials to a disposal facility. Loaded containers are to be stored at the ISA for a period of up to 40 years.

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[1] TRIGA is a trademark of General Dynamics Corporation.

[2] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.

[3] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.

[4] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

D2.1 INTRODUCTION

This chapter was written to satisfy the safety analysis report requirements of U.S. Department of Energy (DOE) Order 5480.23, *Nuclear Safety Analysis Reports*, paragraphs 8.b.(3)(d), as amplified in Attachment 1, paragraphs 4.f.(3)(d)4 a, and 4.f.(8)(b) and (c)2 of the Order, the format and content guidelines of DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, and complies with Title 10, *Code of Federal Regulations* (CFR), Part 830, "Nuclear Safety Management" (10 CFR 830). Facility and component descriptions are provided in support of assumptions used in the hazard analysis and accident analysis discussed in Chapter D3.0. This chapter provides summary level facility and storage system descriptions. Additional information at the level of functional requirements and performance evaluation of the safety structures, systems, and components (SSCs) is provided in Chapter D4.0.

Consistent with DOE-STD-3009-94 guidance, descriptions in this chapter provide a model of the facility systems necessary to develop an understanding of the structures, equipment, and operations without extensive consultation of more controlled design references.

The following topics are discussed in this chapter to the level of detail specified by the precepts of the graded approach described in DOE Order 5480.23:

1. Overview of the facility (Section D2.3)

2. Description of the facility structure and design basis (Section D2.4)

3. Description of the facility process systems and components, and relationships of the SSCs (Section D2.5)

4. Description of confinement systems (Section D2.6)

5. Description of facility safety support systems (Section D2.7)

6. Description of facility electrical systems (Section D2.8)

7. Description of facility auxiliary systems and support facilities (Section D2.9).

D2.2 REQUIREMENTS

D2.2.1 U.S. Department of Energy Regulations, Orders, and Standards

The following DOE Orders, regulations, and standards are applicable to the safety basis of the facility:

- 10 CFR 830, Subpart A, "Quality Assurance Requirements." This rule requires that a sufficient quality assurance program be in place.

- Title 10, *Code of Federal Regulations*, Section 835, "Occupational Radiation Protection" (10 CFR 835). This rule provides requirements for radiation protection programs.
- DOE Order 5480.23, *Nuclear Safety Analysis Reports*. This order provides nuclear safety analysis report content requirements.
- DOE Order 6430.1A, *General Design Criteria*. This order for nonreactor nuclear facilities presents the main reference standards and guides for facility design. In addition, Division 13, "Special Facilities," Section 1300, "General Requirements," and Section 1320, "Irradiated Fissile Material Storage Facilities," requirements are imposed for the 200 Area ISA.
- DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*. This standard is used to determine hazard categories for nuclear facilities.
- DOE Order 5480.28, *Natural Phenomena Hazards Mitigation*. This order is used to define design requirements for seismic events and straight wind.
- DOE-STD-1020-94, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*. This standard provides natural phenomena hazard design and evaluation criteria. The 200 Area ISA was designed and evaluated for seismic events and straight wind in accordance with this standard.
- DOE-STD-1021-93, *Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems, and Components*. This standard is used to define the specific performance category for SSCs.

D2.2.2 U.S. Nuclear Regulatory Commission Requirements and Guidance

In Letter 95-SFD-167, *Implementation of K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy* (Sellers 1995), DOE established the requirement for new SNF Project facilities to achieve "nuclear safety equivalency" to comparable U.S. Nuclear Regulatory Commission (NRC)-licensed facilities. The SNF Project identified the NRC requirements that were needed in addition to existing and applicable DOE requirements to establish nuclear safety equivalency. These NRC requirements and the process used to identify them are documented in HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements*, and in WHC-SD-SNF-DB-009, *Canister Storage Building Natural Phenomena Hazards*. Title 10, *Code of Federal Regulations*, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Section 10 CFR 72.3, "Definitions," defines SSCs that are considered "important to safety." 10 CFR 72.122, "Overall Requirements," requires that the design bases for SSCs important to safety reflect appropriate combinations of effects on normal and accident conditions and the effects of natural phenomena.

For the 200 Area ISA, important-to-safety SSCs have been identified based on the analysis performed in Chapter D3.0 and in accordance with 10 CFR 72.3. Once SSCs have been identified as having a function meeting the definition of important to safety, the requirements for

SSCs important to safety specified in 10 CFR 72 are imposed. A graded approach is applied to an SSC important to safety by using the guidance provided in NUREG/CR-6407, *Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety*. This classification process is described in more detail in Section D4.2.

D2.2.3 Other Codes and Standards

Appendix A of HNF-SD-SNF-PMP-018, *Site-Wide Spent Nuclear Fuel Project Management Plan*, summarizes the requirements of the 200 Area ISA. The 200 Area ISA Standards/Requirements Identification Document identifies federal, state, and local regulations and laws with applicability to the 200 Area ISA. These requirements provide the basis for the design, development, and testing of equipment to safely store SNF at the 200 Area ISA.

DOE/RL-92-36, *Hanford Site Hoisting and Rigging Manual*, provides the requirements for lifting and rigging equipment. All lifts of SNF or packages containing SNF are considered critical lifts.

The documents identified in this section establish the design requirements for the 200 Area ISA.

D2.3 FACILITY OVERVIEW

The ISA is located within the Hanford Site 200 East Area. Its purpose is to provide the location for aboveground dry cask storage of SNF. The 200 Area ISA will provide safe outside storage of the SNF, while protecting fuel integrity through the use of storage systems resistant to natural phenomena hazards. While the majority of the fuel to be stored within the ISA will consist of FFTF SNF, the ISA will also store other SNF from the Hanford Site, including NRF TRIGA and Material Characterization Center commercial LWR fuel.

An equipment storage building (pre-engineered metal building) is proposed near the 200 Area ISA to house lifting devices, impact limiters, transfer casks, and various other types of equipment associated with storage, movement, or transport of casks. This storage building will be located outside the 200 Area ISA fence and provides no safety-related function.

An aboveground dry cask storage location (the ISA) is necessary for the spent fuel because the current storage facilities are being shut down and deactivated. The spent fuel will be transferred to interim storage because there is no permanent repository storage currently available.

The 200 Area ISA is located west of the CSB. The footprint of the ISA is nominally 500 ft by 400 ft surrounded by a 7-ft tall chain-link fence topped with three strands of barbed wire. Five manual gates in the fence, four 30-ft sliding and 1 personnel gate, control access of vehicles and personnel to the 200 Area ISA. Light poles within the perimeter provide illumination. Within the fenced area are three concrete pads, two for placement of ISCs and one for the NAC-1 casks within ISO containers. The Rad-Vault holding NRF TRIGA casks and DOT-6M containers will be placed on graded, compacted gravel.

The two pads for ISC storage are approximately 171 ft by 26 ft by 1.5 ft thick constructed of reinforced concrete. The NAC-1 pad is approximately 88 ft by 28 ft by 1 ft thick constructed of reinforced concrete. This smaller pad runs parallel to the two larger ones.

The three pads are laid out along their length on the north-south axis. The two ISC pads are separated nominally by 43 ft, and the ISO/NAC-1 pad is separated by 80 ft from the nearest ISC pad.

The level gravel area for the Rad-Vault is located northwest of the NAC-1 storage pad. A representation of the facility and its relation to the CSB are provided in Figure D2-1.

The 200 Area ISA facility components are classified as general service; the TRIGA storage system components are also designated general service, while the FFTF and LWR storage systems are designated safety significant. The core component container (CCC) and the NAC-1 canister are designated safety class and have both safety-class and safety-significant functions.

The ISA is surrounded by a galvanized steel chain-link fence at least 7 ft high to restrict personnel access. Access to the facility is provided via four manually operated truck gates and one personnel gate. 10 CFR 72, Sections 72.180, 72.182, 72.184, and 72.186 require physical security plans and system designs in accordance with Title 10, *Code of Federal Regulations*, Part 73, "Physical Protection of Plants and Materials" (10 CFR 73). Security aspects are not addressed in this final safety analysis report.

Sodium lamps mounted on poles located inside the 200 Area ISA at the fence line are provided for illumination of the facility. There are eight light poles located two per side of the fenced enclosure.

No uncontained radioactive materials will be handled at the storage area. Therefore, decontamination and decommissioning efforts should be minimal.

There is interaction of the ISA with existing CSB facilities for surveillance activities and central alarm notification.

Table D2-1 summarizes the specific casks and containers to be used at the 200 Area ISA. The NAC-1 cask generically refers to either the NAC-1 or NFS-4[5] SNF shipping casks that will satisfy both onsite transportation and storage requirements. The NAC-1 and NFS-4 casks were fabricated to the same design drawing, but at different times by different corporate owners. Both model casks will be referred to as the NAC-1 cask throughout this document.

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5 NFS-4 casks were manufactured by Nuclear Fuel Services.

Table D2-1. 200 Area Interim Storage Area Cask/Container Summary.

Cask/Container	Weight	Dimensions
ISC	114,200 lb (max)	85 in. diameter x 181 in. height
Rad-Vault[a]	81,400 lb (max)	114 in. diameter x 111 in. height
TRIGA[b]	2,013 lb (loaded)	16 in. diameter x 38 in. height
DOT-6M[c]	640 lb (loaded)	23 in. diameter x 70 in. height
ISO	8,650 lb (empty)	8 ft high x 8 ft wide x 20 ft long
	6,750 lb (empty)	6 ft high x 8 ft wide x 20 ft long
NAC-1[d]	47,150 lb (loaded)	50 in. diameter (max) x 214 in. long

[a] TRIGA is a trademark of General Dynamics Corporation.
[b] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.
[c] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.
[d] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

ISC = interim storage cask.
ISO = International Standards Organization.

D2.3.1 200 Area Interim Storage Area Hazards

A final hazard category classification of the facility is presented in Chapter D3.0, with the 200 Area ISA classified as a Hazard Category 2 facility. This classification was made in accordance with DOE-STD-1027-92.

The 200 Area ISA hazard analysis is documented in SNF-4820, *200 Area Interim Storage Area Final Hazard Analysis Report.* The only significant inventory of hazardous material in the 200 Area ISA is the radiological content of the dry cask storage systems. The 200 Area ISA does not house nor will it routinely conduct any chemical processes. Radiological guidelines applicable to the radionuclide inventory were found to be more limiting than the toxicological guidelines for the release of SNF particulate. Other hazardous materials identified in the dry cask storage systems include potentially pyrophoric metals and hydrides, oxidizers, and hydrogen.

SNF-4932, *Fire Hazards Analysis for 200 Area Interim Storage Area*, concluded that no release of radioactive material to the environment resulted from any facility fire scenario.

D2.3.2 200 Area Interim Storage Area Operations Summary

The 200 Area ISA operators, supervisors, and managers will be part of the CSB staff.

Placement of Fast Flux Test Facility Fuel Containers

Receipt and set up of the mobile crane precedes the arrival of the transporter carrying the ISC. After the tie-downs are removed from the ISC, the lifting fixture will be attached to the mobile crane, as shown in Figure D2-2. Once the ISC lifting fixture is securely attached to the ISC lifting lugs, the ISC will be hoisted and removed from the transporter. The ISC will be placed in its assigned storage position on the concrete pad. The environmental cover on top of the ISC will be installed. The transporter used to carry the ISC can then be released.

Placement of the Neutron Radiography Facility TRIGA Cask and DOT-6M Containers in the Rad-Vault

The Rad-Vault, Figure D2-3, is not designed to be moved in a loaded condition or with the lid installed. Therefore, loading of the TRIGA casks (see Figure D2-4 and D2-5) and DOT-6M containers (see Figure D2-6) into the Rad-Vault will take place in the 200 Area ISA. Each Rad-Vault is capable of receiving two DOT-6M containers and six TRIGA casks. The casks and the containers are transported two at a time to the 200 Area ISA. Only a single cask or container is allowed to be moved into the Rad-Vault at a time. The TRIGA cask and the DOT-6M container have no specified loading order. Specific details, drawings, and operational restrictions for cask movement are provided in separate procedures. Empty 55-gallon drums are used as spacers inside the Rad-Vault and swapped out one at a time as loaded fuel casks are placed into the Rad-Vault. Final loading will have six TRIGA casks, two DOT-6M containers, and one spacer drum (located in the center position), as shown in Figure D2-7. The spacer drums are only used for ease of loading and provide a passive function of providing a close-packed array such that the containers cannot tip over during or after placement.

Prior to the receipt of the transporter carrying the Rad-Vault body, the mobile crane will be positioned and set up. Once the transporter carrying the Rad-Vault lid has been received, the Rad-Vault body can be placed at the assigned storage location. The Rad-Vault lid is placed adjacent to the Rad-Vault body. The transporters used to carry the Rad-Vault and lid can then be released.

Reception of the transporter carrying two DOT-6M containers can proceed. Using a 55-gallon drum lifter, a spacer drum will be sequentially removed from the Rad-Vault and replaced with a loaded DOT-6M container. This process will be repeated for the second DOT-6M container. The transporter that carried the DOT-6M containers can then be released.

A similar substitution process is used for loading the Rad-Vault with the NRF TRIGA casks. Once the mobile crane is positioned and set up, the reception of the transporter carrying two TRIGA casks in impact limiters will be permitted. Attachment of the cask lift beam assembly to the crane will occur after removing the upper impact limiter from one of the TRIGA casks. The TRIGA cask lift beam assembly will be connected to the TRIGA cask and the cask removed from the lower impact limiter. The cask will then be placed in the Rad-Vault. This process will be repeated for the remaining TRIGA casks. Once the Rad-Vault is loaded, the upper impact limiters will be placed on the empty lower impact limiter located on the transporter, and the transporter used to carry the TRIGA cask can be released. The lid is then placed on the Rad-Vault.

Placement of Commercial Light Water Reactor Fuel Containers

Operations at the 200 Area ISA start with the receipt and set up of the mobile crane and the receipt and staging of the lifting device for the ISO containers. Subsequent to arrival of the transporter carrying a NAC-1 cask in an ISO container, the lifting device is attached to the mobile crane and then attached to the ISO container. The ISO container is then lifted from the transporter and placed on the assigned position on the storage pad. The transporter can then be released.

Surveillance and Maintenance

There are maintenance activities necessary to ensure long-term storage of the fuel at the ISA (e.g., preventative maintenance including painting of storage cask systems). Minimal, non-safety-related maintenance activities will include periodic inspection of the fence, lighting replacement, and repair, if required. Periodic weed removal is performed to minimize the potential for a fire at the pad.

Interim Storage Cask

Periodic inspection of the ISCs is required to support analysis assumptions. Annual surveillance of the ISC includes a visual inspection, radiation survey, and smear sampling about the ISC environmental cover.

Rad-Vault, Neutron Radiography Facility TRIGA Cask, and DOT-6M Container

Annual surveillance of the TRIGA cask and DOT-6M containers shall include (1) visual inspection and radiation surveys of the Rad-Vault, (2) vault lid removal to allow visual inspection and radiation surveys of the fuel casks/containers, and (3) smear sampling of the fuel casks/containers and subsequent vault lid placement. As the Rad-Vault is constructed of concrete, an annual visual inspection of the underside is not required.

NAC-1 Casks

Annual surveillance of the NAC-1 casks in ISO containers shall include (1) a visual inspection of the external surface of the ISO container, (2) radiation surveys of the exterior surface of the ISO, (3) a radiation survey of the upper end of the NAC-1 casks upon opening the ISO container doors, (4) smear samples from the top of the NAC-1 casks, and (5) visual inspections of the interior of the ISO container and exteriors of the NAC-1 casks. As the ISO has structural members constructed of ferrous material, an inspection of the underside will be performed every 5 years. This inspection will be performed in accordance with the requirements of Section D10.4. Operating procedures/work plan will be approved for this activity at that time. Critical lift procedures will be imposed.

D2.3.3 200 Area Interim Storage Area Confinement

Confinement of radioactive material at the 200 Area ISA is a design feature of each spent fuel storage system. These features are summarized below, with additional details provided in Section D2.6 and Chapter D4.0.

Fast Flux Test Facility Fuel

The ISC is an aboveground concrete and steel shielded, top-loading spent fuel storage cask that is used to provide safe interim dry storage of a CCC (see Figures D2-8 and D2-9) loaded with intact FFTF spent fuel assemblies or pin containers. One CCC can be stored in the cavity of each ISC. The ISC design consists of an all-stainless steel internal confinement structure surrounded by steel and concrete shielding. The fully loaded cask weighs a maximum of 114,200 lb, including a loaded CCC with a gross payload of 5,000 lb, the closure hardware, and the weather cover. Outer cask dimensions are 85 in. in diameter and 181 in. tall. The internal cavity of the ISC is 21 in. in diameter and 147 in. tall.

The ISC has been designed and fabricated to meet the requirements of WHC-S-4110, *Specification for FFTF Interim Storage Cask*, in accordance with 10 CFR 72. "Canning" of the spent fuel is provided by the CCC, as discussed in Section D2.6.2.1. The ISC is designed to provide confinement for the fuel, passive heat removal, and environmental protection for the CCC. It also provides radiological shielding protection for site personnel by limiting the dose rate to acceptable levels at normally accessible surfaces. A gasketed weather protection cover is installed on each ISC in the ISA. An additional cover plate may be seal welded over the bolted closure plug after receipt at the 200 Area ISA to enhance the long-term storage configuration.

Neutron Radiography Facility TRIGA Fuel

NRF TRIGA fuel from the NRF in the 300 Area is stored in NRF TRIGA casks and DOT-6M containers. Each NRF TRIGA cask, with capacity to hold up to 18 elements, is nominally 38 in. tall and 16 in. in diameter. Each DOT-6M container, holding one fuel follower control rod (FFCR) element in an inner 2R container, is nominally 70 in. tall and 23 in. in diameter.

Six NRF TRIGA casks and two DOT-6M containers are placed within a concrete Rad-Vault. The Rad-Vault provides environmental protection, supplemental shielding, and natural phenomena hazards resistance. The Rad-Vault is a concrete, vertical, right circular cylinder with light steel reinforcement. The empty Rad-Vault weighs 63,400 lb, consisting of the 43,400-lb. body and 20,000-lb. lid. The loaded weight of the storage system is approximately 81,400 lb. With the lid installed, the Rad-Vault is 111 in. tall and the outer diameter is 114 in.

Commercial Light Water Reactor Fuel

NAC-1 casks within ISO containers are used to store the commercial LWR SNF retrieved from storage at the 324 Building in the Hanford Site 300 Area. Within each NAC-1 cask, a welded inner container provides confinement for the commercial LWR SNF. Each inner container holds either an individual assembly or consolidated pins. The NAC-1 cask is a metal cask that provides structural protection and shielding for the canister. Each NAC-1 cask is approximated by a right circular cylinder 214 in. long, with a maximum diameter of 50 in. Loaded gross weight of the cask is approximately 49,000 lb, excluding the weight of the ISO container. The ISO container provides weather protection and has a footprint of 8 ft by 20 ft.

D2.3.4 200 Area Interim Storage Area Systems

This section is not applicable as there are no mechanical or process systems provided by the ISA. Lifting is provided by a portable crane.

D2.4 FACILITY STRUCTURE

The 200 Area ISA is designed to DOE Order 6430.1A requirements. It is designated as a nonreactor facility under Section 1300, "Special Facilities," which includes consideration of the design requirements under Section 1320, "Irradiated Fissile Material Storage and Handling Facility." The 200 Area ISA is a temporary facility with a design life of 40 years.

This section contains the structural descriptions for the 200 Area ISA and includes the layout for the various storage pads.

D2.4.1 Interim Storage Area Yard

The 200 Area ISA footprint is nominally 500 ft by 400 ft surrounded by a 7-ft tall chain-link fence, with gates in the fence that control access of vehicles and personnel. The ISA is nominally 710 ft above sea level. Light poles around the perimeter provide illumination. Within the fenced area are three concrete pads, two for placement of ISCs and one for placement of the NAC-1 casks within ISO containers. The Rad-Vault holding NRF TRIGA casks and DOT-6M containers will be placed on graded, compacted gravel. All concrete pads have embedded conduit that is intended to support any future monitoring needs.

The fence is no closer than 142 ft to any edge of the NAC-1/ISO concrete pad and 146 ft to the outer edge of the nearest container storage position on the concrete pad.

Excavation, backfilling, and compacting of building materials are in accordance with American Society for Testing and Materials (ASTM) standards ASTM D-653, *Terminology Relating to Soil, Rock, and Contained Fluids*, ASTM D-1557, *Test Method for Laboratory Compaction Characteristics of Soil Using Modified Effort*, ASTM D-2922, *Test Methods for Density of Soil and Soil-Aggregate in Place by Nuclear Methods (Shallow Depth)*, and ASTM D-3017, *Test Method for Water Content of Soil and Rock in Place by Nuclear Methods (Shallow Depth)*.

Concrete work on the pads is in accordance with the American Concrete Institute (ACI) standards ACI-117, *Tolerances for Concrete Construction and Materials*, ACI-224, *Control of Cracking*, ACI-301, *Specifications for Structural Concrete for Buildings*, ACI-306.1, *Cold Weather Concreting*, ACI-318, *Building Code Requirements for Reinforced Concrete*, and ACI-SP-66, *ACI Detailing Manual*; the ASTM A-615, *Deformed and Plain-Billet Steel Bars for Concrete Reinforcement*, ASTM A-853, *Steel Wire, Carbon, for General Use*, ASTM C-33, *Concrete Aggregates*, ASTM C-94, *Ready Mixed Concrete*, ASTM C-150, *Portland Cement*, ASTM C-260, *Air-Entraining Admixtures for Concrete*, and ASTM C-881, *Epoxy-Resin-Base Bonding Systems for Concrete*; Hanford HNF-PRO-097, *Engineering Design and Evaluation*; National Ready Mix Concrete Association, *Certification of Ready Mixed Concrete Production Facilities*; and the *Uniform Building Code* (ICBO 1994).

Polyvinyl chloride (PVC) externally coated galvanized rigid steel conduit (GRC) meeting the National Electrical Manufacturers Association (NEMA) Standard RN-1, *Polyvinyl Chloride (PVC) Externally Coated Galvanized Rigid Steel Conduit and Intermediate Metal Conduit*, is embedded in the concrete pads. This conduit is listed by the Underwriters Laboratory Electrical Construction Materials directory. The conduits are terminated with a rigid steel coupling and pipe plug, installed flush with the top of the concrete. Conduit ends terminating in vaults or handholds are terminated with ground bushings. Ground bushings are connected in series with a minimum of #12 American Wire Gauge (AWG) bare copper wire to six ground conduits of 0.75 in. diameter.

Several fire hydrants are located outside the ISA fence. The hydrants are rated at approximately 1,800 gal/min at 20 lb/in^2.

D2.4.2 Equipment Storage Building

An equipment storage building (pre-engineered metal building), approximately 200 ft to the northeast of the CSB, may be provided near the 200 Area ISA to house lifting devices, impact limiters, transfer casks, and various other types of equipment associated with storage, movement, or transport of the casks. This storage building would be located outside the 200 Area ISA and provides no safety-related function. The proposed storage building consists of a 50 ft by 85 ft one-story insulated metal panel wall and roof on a steel frame building. The building will have heating and cooling and a dry pipe fire protection system. Truck and personal access doors will be provided. A fire hydrant is located northeast of the proposed building.

D2.4.3 Fast Flux Test Facility Interim Storage Cask Pad

The ISC storage pads comprise two pads approximately 171 ft by 26 ft by 1.5 ft thick, in accordance with Drawing H-2-829293, Sheet 4. These pads are reinforced with upper and lower rebar mats (ASTM A-615, Grade 60). The concrete has a minimum compressive strength of 4,000 lb/in^2. The pads include embedded electrical conduits in the concrete slabs with handholds (Utility Vault Company Model #3030-LA) installed outside the pad area. The length of these two pads is coincidental with the north-south axis. The two ISC concrete pads are separated by approximately 43 ft for safety considerations. The western-most edge of the inner ISC concrete pad and the eastern inner edge of the NAC-1/ISO concrete pad are separated by approximately 80 ft.

Each of the ISC pads is intended to support storage of 30 ISC containers. The storage locations are two-abreast. Each position has a nominal circular contact footprint 7 ft in diameter. There are 11 ft between the center lines of adjacent containers. There is nominally 4 ft between the edges of adjacent storage positions, and 4 ft between each outer edge and the edge of the ISC concrete pad. Concrete vehicle barriers (Washington State Department of Transportation

Type 2) are installed in accordance with Washington State Department of Transportation Standard Plan C-8 around the ISC pads in an approximately 200-ft by 330-ft rectangle located approximately 20 ft to 30 ft inside the perimeter fences on the north, east, and south sides. A representation of the 200 Area ISA pad locations relative to the CSB is provided in Figure D2-1.

D2.4.4 Neutron Radiography Facility TRIGA Rad-Vault Gravel Area

The Rad-Vault will not be placed on a concrete pad; instead, a graded and compacted gravel area will be prepared by the project. The design specifications for this gravel area are provided in Drawing H-2-829293. The gravel will be compacted to a soil-bearing capacity greater than 2,500 lb/ft^2.

D2.4.5 Commercial Light Water Reactor International Standards Organization Container Pad

The third storage pad is approximately 88 ft by 28 ft by 1 ft thick, in accordance with Drawing H-2-829293, Sheet 3. This pad is reinforced with upper and lower rebar mats (ASTM A-615, Grade 60). The concrete has a minimum compressive strength of 4,000 lb/in^2. The pad includes embedded electrical conduits in the concrete slab with handholds (Utility Vault Company Model #3030-LA) installed outside the pad area. The length of this pad is coincidental with the north-south axis.

The NAC-1/ISO container pad is designed to support storage of seven containers. The storage locations are sequential with 4 ft of space between adjacent storage ISO container edges and 4 ft to the edge of the concrete pad. Each position has a nominal footprint of 8 ft x 20 ft.

D2.4.6 Design Basis

With more than one type of dry cask storage system stored within the ISA, specific criteria that each dry cask storage system must address will be provided as separate subsections. This section delineates the generic requirements, limits, and siting criteria that must be met by all fuel storage systems residing within the ISA independent of the fuel type and specific design.

A summary of each ISA facility design requirement and its justification or reference is presented in Table D2-2. As fuel storage systems are added, their subsection will address the outline of requirements and limits provided in this section. This will ensure that consistency of the SNF Project authorization basis is maintained and will demonstrate each cask system's acceptability for safe, dry spent fuel storage at the 200 Area ISA.

Because the ISA provides only the general-service functions described in Section D2.3, the cask systems are the major components relied on for meeting the 10 CFR 72 requirements for safe storage of spent fuel. Accordingly, each cask storage configuration provides all the necessary confinement, shielding, criticality control, passive heat removal characteristics, and natural phenomena hazards resistance necessary for the specific spent fuel configuration to be stored.

Table D2-2. Principal Design Requirements for 200 Area Interim Storage Area. (4 sheets)

Category	Requirement	Justification/Reference
Radiological Dose Limits	0.05 mr/hr at the fence 60 mr/hr on contact	HNF-5173[a] limit for uncontrolled access by the public. HNF-5173[a] derived limit for 8 hours exposure per year.
Accident Limits	**Onsite** 1 rem (10^{-1} > frequency > 10^{-2}) 10 rem (10^{-2} > frequency > 10^{-4}) 25 rem (10^{-4} > frequency > 10^{-6}) **Offsite** 500 mrem (10^{-1} > frequency > 10^{-2}) 5 rem (10^{-2} > frequency > 10^{-4}) 5 rem (10^{-4} > frequency > 10^{-6})	Sellers 1995 (95-SFD-167)[b]
Seismic	0.26 g ZPA horizontal, 2/3 of horizontal acceleration in the vertical direction	HNF-D-NF-DB-009, Appendix C[c]
Wind	80 mph fastest mile wind Steady state - 70 mph	HNF-SD-SNF-DB-009, Table 1[c]
Wind Missile	15-lb 2x4 missile at 50 mph.	HNF-SD-SNF-DB-009, Section 3.2[c]
Tornado	200 mph resultant wind speed Differential pressure - 0.90 lb/in^2 over 3 seconds	HNF-SD-SNF-DB-009, Table 1[c] HNF-SD-SNF-DB-009, Section 3.3[c]
Tornado Missile	Not applicable (probability < 10^{-6})	HNF-1785[d]
Flood	Not applicable	Analysis shows the 200 Area is above the flood-affected zone (HNF-SD-SNF-DB-009[c])
Rain	Site drainage: 9.2 in. for 6-hour probable maximum precipitation	HNF-SD-SNF-DB-009, Table 1[c]

Table D2-2. Principal Design Requirements for 200 Area Interim Storage Area. (4 sheets)

Category	Requirement	Justification/Reference
Ash	24 lb/ft^2	HNF-SD-SNF-DB-009[c] imposes an additional roof load for ashfall for safety SSCs. Ashfall shall also be addressed as an insulator during heat load calculations.
Snow	20 lb/ft^2	HNF-SD-SNF-DB-009, Table 1[c]
Soil Loads	Soil density of 110 lb/ft^3 is assumed. Soil bearing capacity = 2,500 lb/ft^2 Concrete design of foundations: UBC[f] and ACI-318[g] (include dynamic earth pressures)	 Drawing H-2-829293, "Civil Site Plan Interim Storage Area"[e]
Lightning	Lightning protection shall be provided for the ISA to meet NFPA 780[h] (if the container cannot withstand a direct strike).	HNF-SD-SNF-DB-009, Table 1[c] Lightning protection required by 10 CFR 72,[i] if container cannot withstand a strike.
Fire	Storage systems shall be designed to withstand the 10 CFR 71.73(c)(3)[j] transportation fire. Defined as a 1,475 °F fire fully engulfed for 30 minutes with return to ambient temperature without water quenching/cooling.	200 Area ISA FHA (SNF-4932[k]) considers the fire analysis prepared for 10 CFR 71.73(c)(3)[j] to be bounding for all scenarios. 10 CFR 72[i] also defaults to this fire.
Load Combinations	Load combinations and allowable stresses for live, dead, snow, and normal operating loads shall be applied. Load combinations and allowable stresses for normal operating loads and natural phenomenon loads. Load combinations for original cask construction.	UBC[f] UCRL-15910[l] (replaced by DOE-STD-1020-94[m]) NRC Regulatory Guide 7.6,[n] ASME Code, Section III[o]

Table D2-2. Principal Design Requirements for 200 Area Interim Storage Area. (4 sheets)

Category	Requirement	Justification/Reference
Thermal	Decay heat removal shall be by passive means only.	No operating safety systems are planned for the ISA due to the passive heat removal cask design.
	Maximum surface temperature under most severe environmental conditions shall not exceed 185 °F.	Temperature limit per 10 CFR 71[j] (revised April 1996).
	Fuel temperature shall not exceed design basis temperatures.	General requirement for all fuel storage systems.
	Cask components with specified temperature limits (i.e., seals) shall not exceed design basis temperatures.	General requirement for all fuel storage systems.
	Environmental temperature range from -27 °F to +115 °F.	Canister Storage Building design basis document.
Internal Pressure	The storage system shall be designed to withstand the worst credible internal pressure generation scenario identified during accident analysis. Testing shall be to 125 - 150% of design per the ASME Code.[o]	General requirement to ensure that outdoor environmental conditions, including direct sunlight, are included in design and accident scenarios involving gas pressure events along with 100% rod rupture.
Radiation Source	Isotopic composition of the stored material shall be defined.	Composition shall be determined either by analysis or calculation.
Shielding	Storage configuration shall meet the requirement of 60 mr/hr external surface contact dose rate in the storage configuration.	HNF-5173[a] limit of 500 mrem/yr and ALARA considerations for 8 hr/yr exposure and 10 CFR 835[p] requirements.
Structural	Systems shall be designed to withstand the conditions as required for their designated safety classification.	The safety classification system requires that SSCs survive specified natural phenomenon events. More severe accident events must not result in consequences that exceed the guidelines identified for normal and abnormal events.

Table D2-2. Principal Design Requirements for 200 Area Interim Storage Area. (4 sheets)

Category	Requirement	Justification/Reference
Handling Loads	The storage system and/or components shall be designed to be lifted by crane.	General requirement for the ISA.
	Systems and components shall be designed to withstand a set down load on rigid surfaces at impact velocities of up to 10 ft/min. (~2 g).	This simulates a hard set down by the crane.
	Systems and components shall be designed to withstand the maximum handling drop anticipated.	Systems and components must be analyzed to withstand the maximum handling drop anticipated.
Criticality	$k_{eff} \leq 0.95$	NRC equivalence.

[a] HNF-5173, 2001, *Project Hanford Radiological Control Manual*, Rev. 1, Fluor Hanford, Incorporated, Richland, Washington.

[b] Sellers, E. D., 1995, *Implementation of the K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy*, (Letter 9504163/95-SFD-167 to President, Westinghouse Hanford Company, September 12), U.S. Department of Energy, Richland Operations Office, Richland, Washington.

[c] WHC-SD-SNF-DB-009, 1996, *Canister Storage Building Natural Phenomena Hazards*, Rev. 4, Westinghouse Hanford Company, Richland, Washington.

[d] HNF-1785, 1997, *Probabilistic Risk Analysis Tornado Missile Hazard to 200 Area Interim Storage Area*, Rev. 0, DE&S Hanford, Incorporated, Richland, Washington.

[e] H-2-829293, 2000, "Civil Site Plan Interim Storage Area," Rev. 1, Fluor Hanford, Incorporated, Richland, Washington.

[f] ICBO, 1994, *Uniform Building Code*, International Conference of Building Officials, Whittier, California.

[g] ACI-318, 1995, *Building Code Requirements for Reinforced Concrete*, American Concrete Institute, Farmington Hills, Michigan.

[h] NFPA 780, 1995, *Lightning Protection Systems*, National Fire Protection Association, Quincy, Massachusetts.

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

[j] 10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*.

[k] SNF-4932, 2000, *Fire Hazards Analysis for the 200 Area Interim Storage Area*, Rev. 0, Fluor Hanford, Incorporated, Richland, Washington.

[l] UCRL-15910, 1988, *Design and Evaluation Guidelines for Department of Energy Facilities Subjected to Natural Phenomena Hazards*, Lawrence Livermore National Laboratory, Livermore, California.

[m] DOE-STD-1020-94 (Change Notice 1-1996), 1994, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*, U.S. Department of Energy, Washington, D.C.

[n] NRC Regulatory Guide 7.6, 1978, *Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels*, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.

[o] ASME, 1995, *Boiler and Pressure Vessel Code*, American Society of Mechanical Engineers, New York, New York.

[p] 10 CFR 835, "Occupational Radiation Protection," *Code of Federal Regulations*, as amended.

ALARA	=	as low as reasonably achievable.	NRC	=	U.S. Nuclear Regulatory Commission.
ASME	=	American Society of Mechanical Engineers.	SSC	=	structure, system, and component.
FHA	=	fire hazards analysis.	UBC	=	Uniform Building Code.
ISA	=	Interim Storage Area.	ZPA	=	zero-period acceleration.

D2.5 PROCESS DESCRIPTION

200 Area ISA has no active processing systems. The information provided in this section includes fuel descriptions and parameters.

D2.5.1 Fuel Description

Table D2-3 summarizes the fuel history and fuel specifics for each SNF dry storage system. FFTF fuel and TRIGA fuel have intact cladding. Intact cladding is defined in NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems*, as, "Spent fuel cladding that does not have gross cladding defects." A gross cladding defect is defined as, "A known or suspected cladding condition that results in the fuel not meeting its design-basis criteria for dry cask storage. The cask shielding, criticality, thermal, and radiological design analyses typically assume that the cladding provides sufficient structural integrity to retain the fuel pellets in the fuel assembly geometry for normal and accident conditions...Alternative means, such as canning, will be required for dry cask storage of fuel that does not meet design-basis conditions." Footnotes to this definition further clarify, "...generally require that the fuel have no known or suspected gross cladding breaches to ensure the structural integrity of the fuel...fuel with cladding defects greater than pin holes and hairline cracks are not authorized...." Pool water sample history or fuel assembly rinse solution samples (instead of visual inspection) may be used to demonstrate that a "gross cladding defect" has not occurred.

Although there are no indications of damage to the commercial LWR fuel, analysis has been performed assuming damaged fuel within the welded inner NAC-1 canister. The criticality analysis and radiological evaluation assumed the canister contained damaged fuel. Accident analyses demonstrate the LWR canister will retain confinement and leaktight functions. Visual fuel inspections will be performed during the loading activity to verify actual fuel condition.

D2.5.1.1 Fast Flux Test Facility Fuel.

The purpose of the FFTF was to provide testing capability to satisfy the diverse technology development needs for the advanced reactor programs. The mission included irradiation and evaluation of different types of fuel assemblies and different materials for fuel assembly construction. Also included was direct production of useful materials such as medical isotopes. In December 1993, DOE directed that the FFTF be transitioned to shutdown status. In April 1994, removal of fuel from the reactor began.

There are nominally 374 fueled components at FFTF that will be placed into storage at the 200 Area ISA. Since cessation of FFTF reactor operations in April 1992, the fuel has radiologically decayed and a substantial reduction in associated fission products and noble gases has occurred. As of August 1995, 351 fueled components were below 200 W decay heat; the highest decay heat assembly was 329 W. The average decay heat, considering all fuel components, is 81 W. All assemblies currently are below 250 W decay heat.

All fuel assemblies fall into two general categories; driver fuel assemblies (DFAs) and test assemblies. The primary purpose of the DFAs was to hold the fuel that creates the neutron flux environment to carry out the various FFTF tests. The primary purpose of the test assemblies was to hold the fuel and non-fuel materials being tested. The fueled test assemblies also contributed to the neutron flux within the reactor.

Table D2-3. Summary of Fuel Characteristics.

	Fast Flux Test Facility	TRIGA[a] cask	TRIGA[a] DOT-6M[b]	Commercial Light Water Reactor
Fuel type	FFTF driver fuel and experimental assemblies	TRIGA[a] fuel element	FFCR	PWR and BWR
Maximum enrichment	29.3% Pu[c]	ZrH with 8.0 - 8.5 wt% U - 20% ^{235}U Enriched		PWR 3.6%/BWR 3.06% 16.92 kg ^{235}U maximum
Burnup	150,000 MWd/MTHM[d]	<1%	<1%	PWR 35,000 MWd/MTHM BWR 34,000 MWd/MTHM
Minimum cooling time	Last irradiated in 1992	Last irradiated 1989	Last irradiated 1976	Last irradiated in 1982
Maximum heat	1500 watts	2.7 watts total in the Rad-Vault		12.4 kW
Maximum fuel load per dry storage system	7 DFA, or 6 Ident-69, or 5 Ident-69 and 2 DFAs	18 fuel elements	1 FFCR	1 PWR assembly or 1 consolidated pin container
Condition of fuel	Intact	Intact	Intact	Damaged fuel assumed in analyses

[a] TRIGA is a trademark of General Dynamics Corporation.
[b] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.
[c] Enrichments greater than 29.3% must be analyzed on a case-by-case basis.
[d] Burnups greater than 150,000 MWd/MTHM must be analyzed on a case-by-case basis.

BWR = boiling water reactor.
DFA = driver fuel assemblies.
FFCR = fuel follower control rod.
FFTF = Fast Flux Test Facility.
MTHM = metric ton of heavy metal.
MWd = megawatt day.
PWR = pressurized water reactor.

The test assemblies have been further categorized for the purpose of this document. The test assemblies that are identical to or very similar to the DFAs are categorized as test DFAs. The test assemblies that contain experimental fuel are categorized as test fuel assemblies, even if they also contain some standard driver fuel pins. The remaining test assemblies did not contain fuel.

The following FFTF fuel types have been authorized for storage at the 200 Area ISA (based on fuel previously authorized for storage at the FFTF 400 Area ISA):

- Standard DFAs.

- Driver evaluation, core characterizer assemblies, and run-to-cladding-breach assemblies are also included based on their similarities to a standard DFA.
- Experimental assemblies and Ident-69 pin containers listed in Table D2-4 have been authorized based on the analyses summarized in WHC-SD-FF-RPT-005, *Review of FFTF Fuel Experiments for Storage at ISA*. Table D2-5 shows experimental assemblies not listed in Table D2-4 that have been disassembled and loaded into Ident-69s.

The remaining fuel and test assemblies require further evaluation before they can be accepted at the ISA. An Engineering Change Notice to this document is required to authorize the additional fuel assemblies listed in Table D2-6.

Table D2-4. Experimental Fuel Assemblies and Ident-69 Pin Containers Analyzed for Loading into a Core Component Container/Interim Storage Cask.

Experimental assemblies	Ident-69 pin containers		
AAD-1, 2, 3, 4, 6, 7	ID-69 S/N 1	ID-69 S/N 23	ID-69 S/N 43
ACO-2, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16	ID-69 S/N 2	ID-69 S/N 24	ID-69 S/N 44
AW-1	ID-69 S/N 3	ID-69 S/N 25	ID-69 S/N 45
CRBR-1, 3, 5	ID-69 S/N 5	ID-69 S/N 26	
CV-1	ID-69 S/N 7	ID-69 S/N 27	
D9-3	ID-69 S/N 8	ID-69 S/N 28	
DE-HTD	ID-69 S/N 9	ID-69 S/N 30	
DEA-2	ID-69 S/N 10	ID-69 S/N 31	
DIPRESS	ID-69 S/N 11	ID-69 S/N 32	
FOTA-1, 2	ID-69 S/N 12	ID-69 S/N 33	
GF001, GF002	ID-69 S/N 13	ID-69 S/N 35	
MW-1, 2, 3, 4, 5, 6	ID-69 S/N 14	ID-69 S/N 36	
PO-2, 5	ID-69 S/N 15	ID-69 S/N 37	
RNTT-1	ID-69 S/N 17	ID-69 S/N 38	
RTCB-4	ID-69 S/N 18	ID-69 S/N 39	
SRF-1, 2	ID-69 S/N 21	ID-69 S/N 40	
WF004, WF005	ID-69 S/N 22	ID-69 S/N 42	

Table D2-5. Pin Container Test Pin Contents. (2 sheets)

Ident-69 S/N	Type[a]	Total pins	Original component	Number of pins	Burnup (MWd/ MTHM)	Decay heat (W) 3/06/96
1	69C	125	2073 FO-1[b] 16392 4.2DFA	91 34	46,910 84,680	62.99
2	69	108	8243 DE-1-6 16397 DE-3-1 16454 DEA-1 16491 BUND-1[b] 16502 D9-1	15 1 44 30 18	22,880 61,760 2,860 53,980 78,830	51.48
4	69C		16492 IFR-1[c]	115	76,710	112.32
5	69C	158	8233 DE-3-2 16397 DE-3-1 16426 DE-1-2 16439 DE-4-1 16458 DE-2-lR 16497 C-1[b]	29 26 32 21 23 27	55,180 61,760 21,360 81,570 44,450 88,050	71.22
8	69C	122	2073 FO-1[b] 2155 MFF-lA[b&d]	78 44	46,910 46,060	68.33
12	69C	120	2170 MFA-1[b] 16392 4.2DFA	56 64	119,590 84,680	59.27 29.01
14	69	109	16502 D9-1[b]	109	78,830	66.79
22	69C	144	16497 C-1[b] 16527 DE-9	98 46	88,050 73,040	80.63
23	69C	133	16489 AAD-5[b]	133	73,860	73.23
24	69C	121	2065 PO-1[b] 16489 AAD-5[b]	103 18	88,310 73,860	97.84
25	69C	145	4123 RTCB-7 16502 D9-1[b]	70 75	70,840 78,830	76.61
26	69C	145	16527 DE-9	131	73,040	69.11
27	69C	99	2144 ACO-3[b] 2170 MFA-1[b] 16392 4.2DFA	12 86 1	197,450 119,590 84,680	19.83 91.02 .45
29	69C	51	2172 MBA-1[b]	51	12,290	24.00
30	69C	154	16491 BUND-1[a]	154	53,980	67.76
31	69C	128	4123 RTCB-7 16491 BUND-1	96 32	70,840 53,980	70.70

Table D2-5. Pin Container Test Pin Contents. (2 sheets)

Ident-69 S/N	Type[a]	Total pins	Original component	Number of pins	Burnup (MWd/ MTHM)	Decay heat (W) 3/06/96
32	69C	146	4123 RTCB-7 16497 C-I[a] 16523 D9-4[b]	15 79 52	70,840 88,050 102,360	87.12
33	69C	151	2171 MFA-2[b]	151	127,780	183.26
35	69C	135	16523 D9-4[b]	135	102,360	88.59
37	69C	128	8252 ACO-1[b]	128	93,920	89.59
38	69C	126	16489 AAD-5[b] 16505 D9-2[b]	57 69	73,860 127,360	83.58
39	69C	134	2065 PO-1[b] 2155 MFF-1A[b&d]	49 85	88,310 46,060	82.22
40	69C	126	2144 ACO-3[b]	126	197,450	208.19
42	69	108	16426 DE-1-2 16454 DEA-1	66 42	21,360 2,860	44.69
43	69	109	16454 DEA-1	109	2,860	40.59
44	69C	139	2143 FO-2[b] 8252 ACO-1[b] 16505 D9-2[b]	16 16 107	46,840 93,920 127,360	95.11
45	69C	113	2143 FO-2[b]	113	46,840	64.16
46	69C	72	2071 AC-3[b]	72	66,690	101.99
X	x	x	2069 PO-4[e]	x	80,000	126.00
X	x	x	2111 ACN – 1[e]	x	67,450	89.99

[a] Maximum pins per type: 69 = 109, 69A = 55, 69C = 217.
[b] Experimental fuel pins.
[c] IFR-1 pins are sodium-bonded metal fuel.
[d] The MFF-LA pins stored at the Fast Flux Test Facility are mixed oxide; all of the sodium-bonded pins from this assembly were shipped to Argonne National Laboratory - West.
[e] PO-4 and ACN-1 are currently intact assemblies but will be disassembled and placed in pin containers.

MTHM = metric ton of heavy metal.
MWd = megawatt day.
S/N = serial number.

Table D2-6. Assemblies that Require Further Evaluation.

Fuel Category	Test Assembly Name	Evaluations Required
Sodium bonded metal fuel	MFF-1, MFF-2, MFF-3, MFF-4,MFF-5, MFF-6, IFR-1 (Ident-69 #4), MFF-8A	Radiological accident release Shielding Criticality (MFF-2 through MFF-6 and MFF-8A only) Design fuel pin temperature limit
Carbide fuel	FC-1, AC-3, ACN-1	Radiological accident release (ACN-1 only) Criticality (ACN-1 only) Thermal (ACN-1 if the assembly remains intact or partially intact [individual pins are acceptable]) Shielding Design fuel pin temperature limit
Delayed neutron monitor signal/encapsulated failed pins	ACN-1, PO-4, PB19#6C (from DFA-16392)	Evaluation of cleaning method and acceptability for storage
Highly enriched fuel	SRF-3, SRF-4	Criticality
Blanket assemblies	ABA-1, ABA-2, ABA-3, ABA-4, ABA-5, ABA-6, WBA-40, WBA-41, MBA-1 (Ident-69 #29 and pin basket PB19C), UO-I, AB-1	Design fuel pin temperature limit Shielding

As noted in Section D6.3.3.1.1, various FFTF experimental pins, cropped fuel pins, and fuel drillings are to be returned to FFTF. Fuel debris is not currently authorized for storage at the 200 Area ISA, as FFTF fuel received at the ISA is required to have intact cladding upon receipt. Experimental fuel debris has been shown acceptable with respect to criticality safety in Chapter D6.0, even though it is not currently authorized for storage at the ISA. However, impacts of fuel debris on radiological accident releases must still be evaluated.

There are 210 DFAs, 65 test DFAs, and 54 test fuel assemblies irradiated at the FFTF. Of these, 1 DFA, 14 test DFAs, and 13 test fuel assemblies have been disassembled. Also, there were 55 DFAs, 1 test DFA, and 2 test fuel assemblies that had been fabricated but not irradiated.

Unirradiated FFTF fuel is not permitted to be stored at the 200 Area ISA for safeguards consideration.

There are 8.6 metric tons of uranium and 2.4 metric tons of plutonium in the FFTF fuel, as shown in Table D2-7. The plutonium and uranium content of the individual assemblies varies depending on the assembly exposure and its pre-irradiation content.

Table D2-7. Fast Flux Test Facility Fuel Types.

Fuel Type	**Uranium (MT)**	**Plutonium (MT)**	**Total (MT)**
Driver Fuel Assemblies	5.2	1.6	6.8
Test Driver Fuel Assemblies	1.7	0.5	2.2
Test Fuel Assemblies	1.7	0.3	2.0
Total	8.6	2.4	11.0

MT = metric ton.

The driver fuel is plutonium-uranium mixed oxide enriched in fissile plutonium (239 and 241). The essential design features that affect performance are as follows:

Fuel column length:	36 in.
Cladding outside diameter:	0.230 in.
Cladding thickness:	0.015 in.
Beginning of life diametrical gap/pellet diameter:	0.0055/0.1945 in.
Smear density:	85.5% theoretical density
Fill gas composition and pressure:	helium (plus tag gas), 1 atmosphere at room temperature

The fuel pins contain mixed uranium/plutonium oxide clad in 20% cold-worked Type 316 stainless steel and are wire wrapped to maintain pin spacing. The 217 pins are oriented on a triangular pitch in the subassembly with a spacer wire 0.056 in. in diameter.

Driver Fuel Assembly

The DFAs are hexagonally shaped components that are composed of 217 fuel pins, a surrounding duct, a shield orifice assembly, an inlet nozzle, load pads, and a handling socket. The assembly is 12 ft long, 4.575 in. wide across the hexagon flats, 5.16 in. wide across the hexagon points, and weighs 381 lb.

The principal structural component is the Type 316 stainless steel, hexagonal duct that extends from a handling socket at the top, to a shield/orifice region located below the fuel pins. The duct wall thickness is nominally 0.120 in., but varies from 0.065 in. through 0.190 in. along the length to facilitate lateral positioning of assemblies within the core and the distribution of force among the assemblies within the core.

The fuel pins are 0.23 in. in diameter, approximately 93.5 in. long, and have a 36-in. fuel bearing region that is centered 65.5 in. from the bottom end of the fuel assembly. Each fuel pin is helically wrapped with a 0.056-in. diameter steel wire to provide lateral spacing along its length. The fuel region contains approximately 150 pressed and sintered, mixed uranium-plutonium oxide pellets. Enrichment details are shown in Table D2-3.

Test Driver Fuel Assemblies

The test DFAs are either identical or very similar to the standard DFA. Each contains fuel with only one of the standard four plutonium enrichments used in the DFA. Some test DFAs have components made of D9 or HT9 stainless steels versus Type 316 stainless steel.

Fuel pins were removed from some test DFAs and shipped offsite for examination. Some of these removed fuel pins were returned whole in Ident-69 containers or as pieces in a cask. Specific details are provided in WHC-SD-SNF-TI-001, *Hanford Spent Fuel Inventory Baseline*. Only intact pins will be stored in Indent-69 containers.

D2.5.1.2 Neutron Radiography Facility TRIGA Fuel.

A 250 kW TRIGA experimental research reactor was operated in the 300 Area intermittently from the late 1970s until its last power run in May 1989. The reactor was manufactured by Gulf General Corporation of San Diego, California, and was used primarily for neutron radiography of FFTF fuel elements and test assemblies. The fuel from the reactor core/pool storage has been removed as part of the decommissioning of the facility. The NRF TRIGA irradiated fuel inventory consists of 99 TRIGA fuel elements and two FFCRs. The standard fuel elements are stored in the NRF TRIGA casks, and the FFCRs are stored in DOT-6M containers. The NRF TRIGA casks and the DOT-6M containers are stored in a Rad-Vault, as described in Section D2.3.2. The fuel elements are clad with either aluminum or Type 304 stainless steel. The FFCRs are clad with Type 304 stainless steel.

There are a total of 66 aluminum-clad elements. Fifty-six elements were used in the NRF TRIGA core for various periods of time during the thirteen years of operation, one was rejected in the first year because of dimensional changes (bowing), and nine were never used in the NRF TRIGA core. All of these aluminum clad fuel elements were received previously irradiated. There were 33 fuel elements received from the Armed Forces Fuel Research Institute Reactor and 33 from the Nevada Test Site.

There are a total of 33 stainless steel clad fuel elements. Twelve elements were new from General Atomics and were used in the core for the entire thirteen years. Twenty elements were received from the deactivated Diamond Ordnance Reactor Facility; however, none of these elements were used in the 300 Area reactor core. One element was an instrumented fuel element from Oregon State University, also never used in the 300 Area reactor core. This instrumented fuel element had a thermocouple tube attached at the upper end. This 8-in. long tubing was bent to 90° with a 0.25 in. radius so it would not interfere with the closure lid of the inner container.

There are two stainless steel clad FFCRs that also were previously irradiated at the Diamond Ordnance Reactor Facility reactor but never used in the 300 Area reactor core.

The design parameters of the TRIGA fuel are summarized in Table D2-8. All the fuel elements are 28.37 in. long and 1.47 in. in diameter. The aluminum-clad elements have a fuel core length of 14 in., while the stainless steel clad elements have a core length of 15 in. The FFCRs are 45.75 in. long and 1.375 in. in diameter. The fuel length for the FFCR is 15 in. The FFCR also includes a borated graphite column above the fuel core that is 14.24 in. long.

The fuel columns in all fuel assemblies are ceramic zirconium hydride with 8 to 8.5 wt% uranium that is nominally 20 percent enriched ^{235}U dispersed throughout the hydride. The ^{235}U content of the fuel assemblies before irradiation ranged from 32 to 40.5 g each. The FFCR typically contained 32 g ^{235}U each. Table D2-9 lists each fuel element and FFCR, including serial number, operating hours in the 300 Area TRIGA reactor, kWh (exposure), fuel source, date received, cladding type, fuel column length, and gram quantity of ^{235}U. The estimated dose rates, based on reported comparisons with underwater measurements in the NRF TRIGA pool, are for single TRIGA fuel elements at midplane. The irradiated fuel elements have limited burn-up from use in the 300 Area TRIGA reactor; the ^{235}U burn-up is probably less than one percent (30 to 35 g) of the total grams of ^{235}U in all of the fuel elements.

The low burn-up has produced only a small amount of fission products. The relatively long period since shutdown (May 1989) has allowed all of the halogens and most of the noble gas inventory to decay.

The two FFCRs are irradiated and also have less than one percent burn-up. They were last irradiated in 1976 and most of the high activity, short-lived fission products have decayed.

The source term was developed using the ORIGEN2 code, which is documented in WHC-SD-TP-ANAL-001, *ORIGEN2 Source Term Calculations for the Neutron Radiography Facility TRIGA Reactor*. The hypothetical maximum irradiated single fuel element radiation source term is 9.1 Ci, assuming a minimum of six years decay from last irradiation. The total radionuclide inventory of 99 irradiated fuel elements and the two FFCRs is estimated to be a maximum of 920 Ci of radioactive material, including less than 7.5 g of plutonium based on the maximum element. The maximum total decay heat rate of the 99 fuel elements and 2 FFCRs is 2.7 W. The thermal source for a single TRIGA fuel element is 2.51×10^{-2} W, with a design capacity of 18 elements per cask yielding a total of 0.45 W per cask (WHC-SD-TP-ANAL-001, Part B, Chapter 8.0). There was no previous operating history provided with the used fuel elements from the Armed Forces Fuel Research Institute, Diamond Ordnance Reactor Facility, or the Nevada Test Site. Operating history has been estimated and correlated to actual direct dose rate readings. Based on this characterization effort, estimated exposures were normalized and ORIGEN2 runs generated for all fuels, as discussed in WHC-SD-SNF-ANAL-010, *ORIGEN2 Calculation Supporting TRIGA Irradiated Fuel Data Package*.

The TRIGA pool water quality was maintained with a purification system. Impurities and minerals were removed to inhibit corrosion or filming. The purification system was carefully monitored and recorded in a weekly logbook. The pool water was sampled monthly and tested by analytical chemistry to determine conductivity and pH balance and to verify that no fission products were present in the water, demonstrating fuel cladding integrity.

Table D2-8. Principal Design Parameters for Neutron Radiography Facility TRIGA Fuel.

Design parameter		Stainless steel clad	Aluminum clad	Fuel follower control rod
Fuel moderated material		$UZrH_{1.6}$	$UZrH_{1.0}$	$UZrH_{1.6}$
Uranium content		8.5 wt%	8.0 wt%	8.5 wt%
Uranium enrichment		~20% ^{235}U	~20% ^{235}U	~20% ^{235}U
Shape of fuel core		Cylindrical	Cylindrical	Cylindrical
Length of fuel core		15 in.	14 in.	15 in.
Diameter of fuel core (OD)		1.43 in.	1.41 in.	1.335 in.
^{235}U content (nominal)		38 g	37 g	32 g
Fuel mixture (atomic ratio)	Zr	1.0	1.0	1.0
	H	1.6	1.0	1.6
	U (wt%)	8.5	8.0	8.5
Zirconium hydride content		91.5 wt%	92.0 wt%	91.5 wt%
Length of top and bottom graphite reflectors		3.47 in.	4.00 in.	14.25 in. (length of borated graphite column)
Cladding material		Type 304 SS	Al-1100, anodized	Type 304 SS
Top and bottom end fixture material		Type 304 SS	Al-1100, anodized	Type 304 SS
Cladding thickness		0.020 in.	0.030 in.	0.020 in.
Diameter (OD)		1.47 in.	1.47 in.	1.375 in.
Overall length		28.37 in.	28.37 in.	45.75 in.
Number of elements		33	66	2

Note: TRIGA is a trademark of General Dynamics Corporation.

OD = outer dimension.
SS = stainless steel.

Table D2-9. Neutron Radiography Facility TRIGA Fuel. (4 sheets)

Fuel element serial	Operation core (kilowatt hours)	NRF Core operating hours	Received from	Date received	Fuel element cladding	Fuel column length (in.)	Original ^{235}U content (g)
2463	0.000	0.000	AFFRI	7/74	AL	14	39.93
2474	0.000	0.000	AFFRI	8/74	AL	14	38.44
2477	0.000	0.000	AFFRI	7/74	AL	14	39.81
2558	0.000	0.000	AFFRI	7/74	AL	14	39.22
2561	0.000	0.000	AFFRI	7/74	AL	14	38.87
2566	0.000	0.000	AFFRI	7/74	AL	14	39.12
2591	0.000	0.000	AFFRI	7/74	AL	14	40.34
2336	185,434.960	806.760	AFFRI	8/74	AL	14	39.38
3077	211,065.867	1,106.259	AFFRI	7/74	AL	14	36.82
2610	316,012.072	1,648.169	AFFRI	7/74	AL	14	39.69
3073	316,012.072	1,648.169	AFFRI	7/74	AL	14	35.83
2335	352,915.235	1,662.330	AFFRI	7/74	AL	14	38.71
2486	378,546.142	1,961.829	AFFRI	7/74	AL	14	40.07
3071	378,546.142	1,961.829	AFFRI	7/74	AL	14	38.04
2602	463,356.822	2,340.749	AFFRI	7/74	AL	14	38.77
2259	563,981.102	2,768.589	AFFRI	7/74	AL	14	39.27
2286	563,981.102	2,768.589	AFFRI	7/74	AL	14	38.75
2482	563,981.102	2,768.589	AFFRI	7/74	AL	14	38.73
2483	563,981.102	2,768.589	AFFRI	8/74	AL	14	39.69
2484	563,981.102	2,768.589	AFFRI	8/74	AL	14	38.95
2573	563,981.102	2,768.589	AFFRI	7/74	AL	14	39.33
2576	563,981.102	2,768.589	AFFRI	8/74	AL	14	39.27
2579	563,981.102	2,768.589	AFFRI	8/74	AL	14	38.73
2584	563,981.102	2,768.589	AFFRI	8/74	AL	14	39.22
2586	563,981.102	2,768.589	AFFRI	8/74	AL	14	40.47
2600	563,981.102	2,768.589	AFFRI	8/74	AL	14	39.14
2603	563,981.102	2,768.589	AFFRI	7/74	AL	14	38.74
2609	563,981.102	2,768.589	AFFRI	8/74	AL	14	38.78

Table D2-9. Neutron Radiography Facility TRIGA Fuel. (4 sheets)

Fuel element serial	Operation core (kilowatt hours)	NRF Core operating hours	Received from	Date received	Fuel element cladding	Fuel column length (in.)	Original ^{235}U content (g)
2611	563,981.102	2,768.589	AFFRI	7/74	AL	14	38.63
2612	563,981.102	2,768.589	AFFRI	8/74	AL	14	38.53
2613	563,981.102	2,768.589	AFFRI	8/74	AL	14	39.60
2616	563,981.102	2,768.589	AFFRI	8/74	AL	14	39.73
3076	563,981.102	2,768.589	AFFRI	7/74	AL	14	37.60
6312 [a]	0.000	0.000	DORF	5/24/79	SS	15	32.01
6313 [a]	0.000	0.000	DORF	5/24/79	SS	15	32.00
2995	0.000	0.000	DORF	5/24/79	SS	15	35.59
2996	0.000	0.000	DORF	5/24/79	SS	15	35.79
2999	0.000	0.000	DORF	5/24/79	SS	15	36.44
3000	0.000	0.000	DORF	5/24/79	SS	15	37.22
3003	0.000	0.000	DORF	5/24/79	SS	15	36.30
3087	0.000	0.000	DORF	5/24/79	SS	15	37.41
3089	0.000	0.000	DORF	5/24/79	SS	15	37.75
3090	0.000	0.000	DORF	5/24/79	SS	15	37.47
3091	0.000	0.000	DORF	5/24/79	SS	15	37.60
3092	0.000	0.000	DORF	5/24/79	SS	15	37.40
3095	0.000	0.000	DORF	5/24/79	SS	15	37.69
3098	0.000	0.000	DORF	5/24/79	SS	15	37.35
3099	0.000	0.000	DORF	5/24/79	SS	15	37.87
3101	0.000	0.000	DORF	5/24/79	SS	15	37.16
3102	0.000	0.000	DORF	5/24/79	SS	15	37.35
3104	0.000	0.000	DORF	5/24/79	SS	15	37.55
3106	0.000	0.000	DORF	5/24/79	SS	15	37.34
3108	0.000	0.000	DORF	5/24/79	SS	15	37.19
3109	0.000	0.000	DORF	5/24/79	SS	15	37.85
3110	0.000	0.000	DORF	5/24/79	SS	15	37.30
3151	0.000	0.000	DORF	5/24/79	SS/TC[b]	15	39.31

Table D2-9. Neutron Radiography Facility TRIGA Fuel. (4 sheets)

Fuel element serial	Operation core (kilowatt hours)	NRF Core operating hours	Received from	Date received	Fuel element cladding	Fuel column length (in.)	Original ^{235}U content (g)
7180	563,981.102	2,768.589	GA	7/74	SS	15	38.99
7181	563,981.102	2,768.589	GA	7/74	SS	15	38.99
7182	563,981.102	2,768.589	GA	7/74	SS	15	38.99
7183	563,981.102	2,768.589	GA	7/74	SS	15	38.00
7184	563,981.102	2,768.589	GA	7/74	SS	15	38.99
7185	563,981.102	2,768.589	GA	7/74	SS	15	38.00
7186	563,981.102	2,768.589	GA	7/74	SS	15	38.99
7187	563,981.102	2,768.589	GA	7/74	SS	15	38.99
7188	563,981.102	2,768.589	GA	7/74	SS	15	38.00
7189	563,981.102	2,768.589	GA	7/74	SS	15	38.99
7190	563,981.102	2,768.589	GA	7/74	SS	15	38.00
7199	563,981.102	2,768.589	GA	7/74	SS	15	38.99
884	100,624.28	427.84	NTS	2/3/76	AL	14	29.24
3065	316,012.072	1,648.169	NTS	7/74	AL	14	36.09
2828	563,981.102	2,768.589	NTS	2/3/76	AL	14	37.14
2831	563,981.102	2,768.589	NTS	2/3/76	AL	14	36.22
2832	563,981.102	2,768.589	NTS	2/3/76	AL	14	36.32
2834	563,981.102	2,768.589	NTS	2/76	AL	14	37.62
2837	563,981.102	2,768.589	NTS	2/3/76	AL	14	36.05
2840	563,981.102	2,768.589	NTS	2/3/76	AL	14	35.92
2843	563,981.102	2,768.589	NTS	2/3/76	AL	14	37.61
2849	563,981.102	2,768.589	NTS	2/3/76	AL	14	35.09
2850	563,981.102	2,768.589	NTS	2/3/76	AL	14	36.33
2851	563,981.102	2,768.589	NTS	2/3/76	AL	14	34.54
2855	563,981.102	2,768.589	NTS	2/3/76	AL	14	36.87
2859	563,981.102	2,768.589	NTS	2/3/76	AL	14	37.62
2860	563,981.102	2,768.589	NTS	2/3/76	AL	14	35.66
2862	563,981.102	2,768.589	NTS	2/3/76	AL	14	38.00

Table D2-9. Neutron Radiography Facility TRIGA Fuel. (4 sheets)

Fuel element serial	Operation core (kilowatt hours)	NRF Core operating hours	Received from	Date received	Fuel element cladding	Fuel column length (in.)	Original ^{235}U content (g)
2864	563,981.102	2,768.589	NTS	2/3/76	AL	14	35.89
2867	563,981.102	2,768.589	NTS	2/3/76	AL	14	36.86
2871	563,981.102	2,768.589	NTS	2/3/76	AL	14	36.75
2874	563,981.102	2,768.589	NTS	2/3/76	AL	14	35.91
2876	563,981.102	2,768.589	NTS	2/3/76	AL	14	35.71
2877	563,981.102	2,768.589	NTS	2/3/76	AL	14	37.72
2883	563,981.102	2,768.589	NTS	2/3/76	AL	14	36.31
2884	563,981.102	2,768.589	NTS	2/3/76	AL	14	37.33
2890	563,981.102	2,768.589	NTS	2/3/76	AL	14	35.90
2891	563,981.102	2,768.589	NTS	2/3/76	AL	14	36.36
2892	563,981.102	2,768.589	NTS	2/3/76	AL	14	35.02
2894	563,981.102	2,768.589	NTS	2/3/76	AL	14	40.10
2895	563,981.102	2,768.589	NTS	2/3/76	AL	14	37.78
2896	563,981.102	2,768.589	NTS	2/3/76	AL	14	36.25
2897	563,981.102	2,768.589	NTS	2/3/76	AL	14	35.97
3067	563,981.102	2,768.589	NTS	7/74	AL	14	37.06
3069	563,981.102	2,768.589	NTS	8/74	AL	14	37.88

Note: TRIGA is a trademark of General Dynamics Corporation.

[a] Fuel-follower control rod.

[b] Assembly included a thermocouple.

AFFRI = Armed Forces Fuel Research Institute.
AL = aluminum.
DORF = Diamond Ordnance Reactor Facility.
GA = General Atomics Corporation.
NRF = Neutron Radiography Facility.
NTS = Nevada Test Site.
SS = stainless steel.

D2.5.1.3 Commercial Light Water Reactor Fuel.

Seven commercial LWR SNF assemblies were delivered to the Pacific Northwest National Laboratory (PNNL) Material Characterization Center at Hanford in 1985 and 1986 in support of DOE spent fuel repository studies. Five were pressurized water reactor (PWR) assemblies and two were boiling water reactor (BWR) assemblies. As part of the DOE spent fuel repository studies, PNNL removed several fuel rods from the assemblies and designated them approved testing materials (ATMs) to support laboratory investigations of nuclear waste disposal forms. As a result, the fuel from these assemblies has been well characterized.

Two of the five PWR assemblies supplied to PNNL contained three designations of spent fuel from the Calvert Cliffs Unit No. 1 PWR operated by Baltimore Gas and Electric, located near Lusby, Maryland. The first designation of fuel consisted of one full assembly containing 176 rods of moderate burn-up fuel and was identified as assembly 1D101. The second designation consisted of a partial assembly containing 135 rods of high burn-up fuel and was identified as assembly 1D047. Assembly 1D047 also contained twenty rods from a third designation of fuel that was taken from assembly BT03 and inserted into assembly 1D047 to facilitate fuel shipment to the PNNL facility.

For the PNNL Material Characterization Center examinations, fuel rods taken from assemblies 1D101, 1D047, and BT03 were designated as ATM-103, ATM-104, and ATM-106, respectively. Assembly 1D101 had eight rods removed, leaving 168 rods remaining in the assembly. Six of the eight removed rods remain as loose rods for consolidation and shipping to storage. Assembly 1D047 had a total of sixteen rods removed, nine from assembly 1D047 fuel and seven of the BT03 rods, leaving a total of 139 rods remaining in the assembly. Seven of the 1D047 rods and four of the BT03 rods remain for consolidation with the other loose rods for shipping and storage.

Fuel from the Point Beach assemblies was never used as sample material, consequently ATM designations were not assigned to this fuel and the assemblies have remained intact with none of the fuel rods removed. The three remaining PWR assemblies were supplied to PNNL from the Point Beach Unit No. 1 PWR operated by Wisconsin Electric Power Company in Wisconsin. These assemblies were received as three complete fuel assemblies, each containing 179 rods, with assembly identification numbers H-07, H-12, and H-25.

The segmented rod program (SRP) segmented rods from General Electric Vallecitos, which are actually quarter-length BWR fuel rods designed to screw together forming rod assemblies for irradiation in rod positions in standard BWR fuel assemblies, were irradiated in Quad Cities I (1-rod assembly) and Monticello (2-rod assemblies). Following destructive examination of six of these rods, one rod from Quad Cities (SRP-1) and five rods from Monticello (SRP-2) remain. Characterization data for the segmented rods includes axial core positions (i.e., top-center, bottom-center, or bottom) during irradiation. Table D2-10 provides a summary of the dimensions of the various fuel assemblies and fuel rods.

Table D2-10. Commercial Light Water Reactor Fuel Physical Parameters. (2 sheets)

Assembly No.	**ATM No.**	**Manuf.**	**Bundle Length**	**Bundle Width**	**Rod Length**	**Pin Diameter ID, OD**	**Active Fuel Length**	**Pellet Length**	**Pellet Diameter**
1D101 (Calvert Cliffs)	103	Combustion Engineering 14x14	157.25 in	8.125 in.	147 in.	0.388 in. ID 0.440 in. OD	136.7 in.	0.45 in.	0.3765 in.
1D047 (Calvert Cliffs)	104	Combustion Engineering 14x14	157.25 in	8.125 in.	147 in.	0.388 in. ID 0.440 in. OD	136.7 in.	0.45 in.	0.3765 in.
BT03 (Calvert Cliffs)	106	Combustion Engineering 14x14	--	--	147 in.	0.388 in. ID 0.440 in. OD	136.7 in.	0.650 in.	0.3795 in.
H-07 (Point Beach)	--	Westinghouse 14x14	159.71 in	7.76 in.	151.85 in.	0.422 in. OD	144 in.	0.600 in.	0.3659 in.
H-12 (Point Beach)	--	Westinghouse 14x14	159.71 in.	7.76 in.	151.85 in.	0.422 in OD	144 in.	0.600 in.	0.3659 in.
H-25 (Point Beach)	--	Westinghouse 14x14	159.71 in.	7.76 in.	151.85 in.	0.422 in. OD	144 in.	0.600 in.	0.3659 in.
CZ 346 (Cooper Station)	105/ 108	General Electric 7x7	--	--	163.8 in.	0.526 in. ID 0.563 in. OD	146 in.	0.5 in.	0.477 in.
CZ 348 (Cooper Station)	108	General Electric 7x7	--	--	163.8 in.	0.526 in. ID 0.563 in. OD	146 in.	0.5 in.	0.477 in.
SRP-1 (Quad Cities) Rod No 1B03-4	--	General Electric	--	--	38.5 in.	0.489 in. ID 0.563 in. OD	38.275 in.	0.5 in.	0.42 in.

Table D2-10. Commercial Light Water Reactor Fuel Physical Parameters. (2 sheets)

Assembly No.	ATM No.	Manuf.	Bundle Length	Bundle Width	Rod Length	Pin Diameter ID, OD	Active Fuel Length	Pellet Length	Pellet Diameter
SRP-2 (Monticello)									
Rod No OA03-2	--	General Electric	--	--	38.41 in.	0.425 in. ID 0.493 in. OD	29.63 in.	0.421 in.	0.42 in.
Rod No OA08-1	--	General Electric	--	--	38.41 in.	0.425 in. ID 0.493 in. OD	29.63 in.	0.421 in.	0.42 in.
Rod No SD17-3	--	General Electric	--	--	38.41 in.	0.425 in. ID 0.493 in. OD	29.63 in.	0.416 in.	0.42 in.
Rod No SD18-2	--	General Electric	--	--	38.41 in.	0.437 in. ID 0.493 in. OD	29.63 in.	0.428 in.	0.42 in.
Rod No 8D14-1	--	General Electric	--	--	38.41 in.	0.437 in. ID 0.493 in. OD	29.63 in.	0.43 in.	0.42 in.

ATM = approved testing material.
ID = inner dimension.
OD = outer dimension.

The BWR fuel rods are from two assemblies from Cooper Nuclear Power Plant in Brownsville, Nebraska. These assemblies, CZ346 and CZ348, were designated as ATM-105, except for 10 rods (five from each assembly) that contain Gadolinia, which were designated as ATM-108. Twelve rods were removed from CZ346, of which three were destructively examined (including one ATM-108 rod), and nine loose rods remain. The remaining 37 rods from CZ346 and the 49 rods from CZ348 will be disassembled from the fuel bundle hardware for consolidation with the loose PWR and SRP rods in a container, which allows for handling and packaging comparable to a fuel assembly.

The five PWR fuel assemblies and two BWR fuel assemblies were delivered to Hanford and received by PNNL from October through December 1985. Subsequently, the assemblies, except for the rods removed for the ATM, have been stored in thimbles in air in a 324 Building hot cell. The removed rods were stored in a different hot cell during the time when the outer surface of the fuel assemblies became contaminated (primarily with cesium) from other hot cell activities. The contaminated assemblies will be sprayed with air prior to removal from the hot cell to remove loose debris. The hot cell environment in which the assemblies became externally contaminated has been evaluated to demonstrate the assemblies can be stored without decontamination (HNF-8897, *324 Building Spent Nuclear Fuel Handling History from Characterization to Packaging for Transfer to the Interim Storage Area (OCRWM)*).

The principal design parameters of the five PWR fuel assemblies, the two BWR assemblies from which the BWR rods are taken, and the SPR rod segments are described in this section. Table D2-11 also summarizes the assemblies and fuel rods, their fuel data, and cask designations.

Six NAC-1 spent fuel transport casks will provide onsite transport and interim storage of the spent fuel. Each of the five PWR fuel assemblies and a rod container consolidating the loose rods will be loaded into a seal-welded canister that will be placed within each NAC-1 cask. The following sections provide the principle design parameters of each fuel assembly by cask designation. The casks have been designated for the 200 Area ISA storage as follows:

- NAC-1/ISA-A 1D101
- NFS-4/ISA-B 1D047
- NAC-1/ISA-C H-07
- NAC-1/ISA-D H-12
- NAC-1/ISA-E H-25
- NFS-4/ISA-F Consolidated rods.

PNNL-11273, UC-812, *Inventory of LWR Spent Nuclear Fuel in 324 Building*, provides specific details concerning the fuel composition.

Table D2-11. Commercial Light Water Reactor Fuel Rods. (2 sheets)

Cask no.	Fuel type	ID no.	No. rods	ATM no.	Burnup MWd/ MTHM	Enrich wt. %[a]	Poison	Mass kg U	Manuf/ type	Clad	Discharge date	Decay heat (watts)
NAC-1/ ISA-A 1D101	PWR Assembly Calvert Cliffs	1D101	168	103	30,700	2.72		372.1	CE 14x14	Zircaloy-4	10/18/80	302.5
NFS-4/ ISA-B 1D047	PWR Assembly Calvert Cliffs	1D047 BT03	126 13	104 106	41,800 42,700	3.04 2.45		279.1 29.1	CE 14x14 CE 14x14	Zircaloy-4 Zircaloy-4	10/18/80 10/18/80	371.5
NAC-1/ ISA-C H-07	PWR Assembly Point Beach	H-07	179		32,300	3.2		401.1	WE 14x14	Zircaloy-4	10/19/81	356.4
NAC-1/ ISA-D H-12	PWR Assembly Point Beach	H-12	179		32,300	3.2		401.1	WE 14x14	Zircaloy-4	10/19/81	356.4
NAC-1/ ISA-E H-25	PWR Assembly Point Beach	H-25	179		32,300	3.2		401.1	WE 14x14	Zircaloy-4	10/19/81	356.4
NFS-4/ ISA-F Consoli-dated	Consolidated <u>Fuel Rods</u>											
	PWR Rods Calvert Cliffs	1D101	6	103	30,700	2.72		13.29	CE 14x14	Zircaloy-4	10/18/80	10.8
		D047	7	104	41,800	3.04		15.50	CE 14x14	Zircaloy-4	10/18/80	18.7
		BT03	4	106	42,700	2.45		8.96	CE 14x14	Zircaloy-4	10/18/80	10.7
	BWR Rods Cooper Station	CZ346	37	105/1	28,000	2.5		143.9	GE 7x7	Zircaloy-2	5/21/82	105.9
		CZ346	9	08	28,000	2.5	Gd	35	GE 7x7	Zircaloy-2	5/21/82	25.8
		CZ348	49	108	27,500	2.5		190.6	GE 7x7	Zircaloy-2	5/21/82	140.2
	BWR Seg Rods Quad Cities 1	SPR-1	1	105/1	22,600	2.56	Hf-Y	0.63	GE Segm	Zircaloy-2	8/30/80	0.7[c]
	BWR Seg Rods Monticello	SPR-2	5	08	23,700[b]	2.87	Hf-Y	3.17	GE Segm	Zircaloy-2	9/1/82	<u>3.6[c]</u>
												316.4

Notes on next page

Table D2-11. Commercial Light Water Reactor Fuel Rods. (2 sheets)

Notes for Table D2-11:

[a] Initial enrichment
[b] Average
[c] Estimated from decay heat per BWR rod.

Information derived from: PNNL-11273, UC-812, 1996, *Inventory of Spent Nuclear Fuel in the 324 Building,* using data based on decay to the year 2000.
PNL-5109-105, UC-802, 1991, *Characterization of Spent Fuel Approval Testing Material – ATM-105.*
GE Special Agreement No. M-5314-A-F, 1985, *Data Package for Fuel Segments.*
WHC-SD-SNF-TA-010, 1996, *Evaluation of Other Spent Nuclear Fuel Management at the Canister Storage Building Complex, Rev. 0.*

ATM = approved testing material.
BWR = boiling water reactor.
CE = Combustion Engineering, Incorporated.
GE = General Electric Company.
MTHM = metric ton heavy metal.
MWd = megawatt day.
PWR = pressurized water reactor.
WE = Westinghouse Electric Corporation.

Cask NAC-1/ISA-A 1D101 (Calvert Cliffs Fuel)

Cask NAC-1/ISA-A 1D101 will contain fuel assembly 1D101. This assembly contains 168 fuel rods of moderate burnup fuel. Assembly 1D101 is a standard Combustion Engineering, Incorporated, 14x14 PWR fuel assembly. The assembly length is 157.25 in. The width of the fuel bundle is 8.125 in., and the fuel rods are 147 in. in length. All of the fuel rods are clad with Zircaloy-4 tubing. Additional fuel dimensional data is provided in Table D2-11.

The fuel rods from assembly 1D101 were irradiated during cycles 2, 3, and 4 of the Calvert Cliffs reactor operation between March 1977 and October 1980. The fuel assembly was discharged from the reactor October 18, 1980, and stored wet in the reactor fuel storage basin until September 1985, when it was shipped dry to PNNL. The fuel contains no burnable poisons and had an initial enrichment of 2.72 wt%. The initial total mass of uranium contained in assembly 1D101 with the actual remaining rods is 372.1 kg U. The burn-up of the assembly is 30,700 megawatt day/metric ton of heavy metal (MWd/MTHM), and the calculated decay heat is 302.5 W.

Cask NFS-4/ISA-B 1D047 (Calvert Cliffs Fuel)

Cask NFS-4/ISA-B 1D047 will contain fuel assembly 1D047. This assembly contains 126 original fuel rods and thirteen fuel rods from another assembly (BT03), for a total of 139 rods. Assembly 1D047 is a standard Combustion Engineering, Incorporated 14x14 PWR fuel assembly. The assembly length is 157.25 in. The width of the fuel bundle is 8.125 in., and the fuel rods are 147 in. in length. All of the fuel rods are clad with Zircaloy-4 tubing. Additional fuel dimensional data is provided in Table D2-11.

The fuel rods from assembly 1D047 were irradiated during cycles 2, 3, 4, and 5 of operation of the Calvert Cliffs reactor between March 1977 and April 1982. The fuel assembly was discharged from the reactor April 17, 1982, and stored wet in the reactor fuel storage basin until September 1985, when it was shipped dry to PNNL. This fuel contains no burnable poisons and had an initial enrichment of 3.04 wt%. The initial total mass of uranium contained in the remaining 1D047 fuel rods (excluding the BT03 rods) is 279.1 kg U. The burn-up of the assembly is 41,800 MWd/MTHM, and the calculated decay heat is 336.7 W.

The fuel rods from assembly BT03 were irradiated during cycles 1, 2, 3, and 4 between October 1974 and October 1980. The fuel assembly was discharged from the reactor October 18, 1980. Thirteen fuel rods from this assembly are contained within assembly 1D047 and will be stored within cask NFS-4/ISA-B 1D047. This fuel contains no burnable poisons and had an initial enrichment 2.45 wt%. The initial total mass of uranium contained in the thirteen BT03 fuel rods is 29.2 kg U. The burn-up of the fuel is 42,700 MWd/MTHM, and the calculated decay heat for the 13 remaining fuel rods is 34.7 W.

The 1D047 fuel rods combined with the BT03 fuel rods in the 1D047 assembly result in 308.2 kg U for the total mass of uranium contained within the assembly. The combined heat load for the assembly results in 371.5 W. This value is the bounding heat load expected from any of the PWR assemblies.

The BT03 fuel contained within the 1D047 assembly is the most recent fuel of the five PWR assemblies to have been irradiated in a reactor. The BT03 rods were last irradiated in 1982. The remaining PWR fuel was discharged from the reactor before this date. During this storage period, unburned uranium and fission by-products from reactor irradiation of the fuel have continued to radioactively decay.

Cask NAC-1/ISA-C H-07 (Point Beach Fuel)

Cask NAC-1/ISA-C H-07 will contain fuel assembly H-07. This assembly contains 179 fuel rods. Assembly H-07 is a standard Westinghouse 14x14 design. The assembly length is 159.8 in. The width of the fuel bundle is 7.76 in., and the fuel rods are 151.85 in. in length. All of the fuel rods are clad with Zircaloy-4 tubing. Additional fuel dimensional data is provided in Table D2-11.

The fuel rods from assembly H-07 were irradiated during cycles 5, 6, 7, 8, and 9 between November 1976 and October 1981. The assembly was discharged from the reactor October 9, 1981. This fuel contains no burnable poisons and has an initial enrichment of 3.19 wt%. The initial total mass of uranium contained in the assembly is 401.1 kg U. The burn-up for assembly H-07 is 32,300 MWd/MTHM, and the calculated decay heat is 356.4 W.

Cask NAC-1/ISA-D H-12 (Point Beach Fuel)

Cask NAC-1/ISA-D H-12 will contain fuel assembly H-12. This assembly is identical in fuel description and irradiation history to assembly H-07. This fuel has an initial enrichment of 3.19 wt%. The initial total mass of uranium contained in the assembly is 401.1 kg U. The burn-up of the fuel is 32,300 MWd/MTHM, and the calculated decay heat is 356.4 W.

Cask NAC-1/ISA-E H-25 (Point Beach Fuel)

Cask NAC-1/ISA-E H-25 will contain fuel assembly H-25. This assembly is identical in fuel description and irradiation history to assemblies H-07 and H-12. This fuel has an initial enrichment of 3.19 wt%. The initial total mass of uranium contained in the assembly is 401.1 kg U. The burn-up of the fuel is 32,300 MWd/MTHM, and the calculated decay heat is 356.4 W.

Cask NFS-4/ISA-F Consolidated (Cooper, Calvert Cliffs, Quad Cities I, and Monticello Fuel)

Cask NFS-4/ISA-F Consolidated will contain 95 163.8-in. rods from two standard, General Electric 7 x 7 BWR assemblies, CZ346 and CZ348; six 147-in. rods from 1D101; seven 147-in. rods from 1D047; four 147-in. rods from BT03; one 38-in. General Electric BWR rod from SRP-1; and five 38-in. General Electric BWR rods from SRP-2. The 1D101, 1D047, and BT03 fuel and irradiation history are described with the contents of Cask NAC-1/ISA-A and Cask NFS-4/ISA-B. Additional fuel dimensional data is provided in Table D2-11.

Fuel assembly CZ346 and CZ348 were irradiated in cycles 1, 2, 3, 6, and 7 of the operation of the Copper Nuclear Power Plant between July 1974 and May 1982. The two assemblies were shipped from the Cooper storage pool to the pool at the General Electric Morris

Facility, used in a dry storage cask test, and shipped to Pacific Northwest Laboratory in January 1986. No adverse effects on fuel rod integrity were found after testing. Nine of the rods contain Gadolinia. The initial enrichment of the fuel was 2.5 wt%. The initial total mass of uranium in the remaining rods is 370 kg U. The burn-up of the fuel is 28,000 MWd/MTHM, and the calculated decay heat is 272 W.

The irradiation history and shipments to PNNL for the PWR rods is included in the descriptions in the above paragraphs. The SRP-1 rods from Quad Cities I reactor were irradiated during cycles 2 through 5, between July 1974 and August 1980, for a burnup of 22,600 MWd/MTHM. SRP-2 rods from Monticello reactor were irradiated during cycles 3 through 8 (three rods) and 3 through 9 (two rods), between May 1974 and April 1981 and between May 1974 and September 1982, respectively. The average burnup is 23,700 MWd/MTHM. The initial total mass of uranium in the SRP-1 is 0.81 kg U, and 3.17 kg U in the five SRP-2 rods. The decay heat of the SRP rods is estimated to be 4.3 W. The combined heat load for this cask is 316.4 W.

D2.6 CONFINEMENT SYSTEMS

10 CFR 72.122(h)(1) requires that (1) confinement barriers and systems be provided such that the spent fuel cladding is protected during storage against degradation that leads to gross ruptures of the fuel, or (2) the fuel must be otherwise confined such that degradation of the fuel during storage does not pose operational safety problems with respect to its removal from storage. This section identifies and describes the functional, physical, and operational features of the confinement systems that provide protection against release of radioactive or hazardous material. Detailed system descriptions are provided in Chapter D4.0.

D2.6.1 Confinement Approach and Configuration

The primary confinement feature for the FFTF and TRIGA fuel dry cask storage systems (with bolted closures) is the intact fuel cladding. The inner or outer storage container provides a second confinement boundary. For the commercial LWR fuel storage, the primary confinement boundary is the welded inner canister. The NAC-1 cask is not credited for confinement during storage. Specific details for each storage system are provided in the subsections that follow.

D2.6.2 Confinement System Descriptions

D2.6.2.1 Fast Flux Test Facility Fuel.

The primary confinement feature is the intact fuel cladding, which serves to ensure cladding equivalency is maintained during the spent fuel dry storage design lifetime. The secondary confinement feature consists of the ISC, which serves as the main pressure boundary and leakage barrier to the environment to ensure that environmental, worker, and public safety is maintained. The ISC design is consistent with commercial spent fuel dry storage practice. This design restricts oxygen and moisture in-leakage to minute levels such that degradation of the spent fuel is expected to be effectively controlled to an inconsequential rate over the 50-year storage lifetime. The CCC provides fuel retrieval capability, if degradation does occur.

Interim Storage Cask Confinement Boundary

The ISC confinement structure consists of the 1.5-in. thick annular wall, an 8-in. thick bottom head, an 8-in. thick top closure plate, a 4-in. thick top flange, two test-port welded covers, and double metal seals between the top closure and the flange. The top closure of the ISC is held in place by sixteen 1.5-in. diameter bolts. The test port through the closure is provided for evacuation and helium backfill of the cask, once the CCC is installed. After the cask is inerted with helium and the main closure seals certified to a leak rate of $< 1 \times 10^{-7}$ scc/sec, the two 0.5-in. thick test port covers are sequentially seal welded in place and dye penetrant tested. The test port also allows sampling of the cask atmosphere, but no such sampling is currently required or anticipated during normal storage operations. The confinement barrier design conforms to the requirements of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (ASME 1995), Section III, Subsection NC, for safety-significant components. Guidance for application of the ASME Code was taken from NRC Regulatory Guide 7.6, *Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels*.

Core Component Container Boundary

The CCC is constructed to "can" the spent fuel and provide a barrier, in addition to the intact FFTF spent fuel cladding. This barrier is provided to ensure retrievability of the spent fuel in the unlikely event of significant cladding degradation. The CCC confinement boundary consists of the seven closed-bottom storage tubes welded to a single upper support plate, which provides the sealing surface, and the bolted cover that has a metallic seal (see Figure D2-9). The metallic seal between the cover and the container is positioned to pass radially inward of the bolt holes and outward of the upper rim of the storage tubes, thus excluding the closure bolts from this boundary. Additional elastomer seals are located at the outer circumference of the outer plate and at each bolt. A 0.375-in. diameter test port is provided to access the space between the metal and elastomer seals for leak testing. Before acceptance from the manufacturer, each CCC was certified to a leak rate of $< 1 \times 10^{-3}$ scc/sec. After loading the CCC with spent fuel in the interim examination and maintenance (IEM) cell, the seal leak rate was verified at $< 1 \times 10^{-1}$ scc/sec. The IEM cell atmosphere was the inerting atmosphere for long-term storage of the spent fuel because the CCC was closed prior to transfer out of the IEM cell. This atmosphere is argon with minor (< 200 ppm) levels of oxygen and water impurities. The potential levels of oxygen and water within the IEM cell were evaluated in WHC-SD-FF-TA-039, *Radiological Consequences of a Hypothetical Disruption of a Maximally Loaded FFTF Fuel Cask*, and determined to have no significant effect on the spent fuel storage. Maintaining an argon atmosphere is not a safety function.

The CCC (Figure D2-9) is an unshielded, sealed fuel storage container with seven fuel storage positions. The CCC provides canning for the fuel to ensure fuel retrievability is maintained during the spent fuel dry storage design lifetime. The CCC also provides the geometry to ensure criticality control during handling and storage of the fuel. The CCC is designed such that it is fully retrievable from the storage configuration, although the capability to remove individual fuel components from a CCC is not required or guaranteed. The center storage location can accept a fuel assembly that has had the bottom 15.5 in. removed. However, due to the indented CCC handling socket, an Ident-69 pin container cannot be stored in the center location. Therefore, the maximum loading of a CCC will be either six Ident-69 pin containers or seven FFTF fuel assemblies. Pin containers and fuel assemblies can be mixed in a CCC with up to five Ident-69 containers and two DFAs.

The CCC is fabricated from stainless steel and nickel alloy material to provide corrosion-resistant fuel storage. The CCC's overall dimensions are 20.0 in. in diameter by 146 in. in height. The weight of the empty CCC is 1,100 lb. The maximum gross weight of a loaded CCC occurs with seven DFAs. The weight of the seven DFAs is 3,900 lb, giving the CCC a maximum gross weight of 5,000 lb.

The upper portion of the outer tubes has a 6.69-in. outside diameter. There is a saddle section 14 in. above the bottom of the CCC, where each outer storage tube transitions to a smaller section measuring 4.00 in. in outside diameter. The bottom 12.4 in. of the lower section is fabricated from nickel alloy material for enhanced corrosion resistance. The center tube has a 6.54-in. outer diameter. The bottom 10.0 in. of the center tube is also fabricated from nickel alloy material. There is no size reduction at the lower end of the center tube.

The outer storage tubes are suspended from the upper support plate. The center storage tube connects the upper support plate with the lower support plate. The outer tubes are fixed only at their upper ends so they are free to accommodate thermal expansion. This design also permits the outer tubes to stretch slightly to absorb energy during a CCC drop accident onto the ISC internal impact limiter. Drop energy is absorbed by the CCC tubes until the gap between the tube stop and the lower support plate is taken up. The CCC then acts as a rigid body for final interface with the ISC impact limiter during an accident event.

The lower support plate has an 18.0-in. diameter and is 1.5 in. thick. This support plate limits downward travel of the outer storage tubes, and also provides radial positioning guidance for inserting the CCC into the ISC. The upper support plate has a 20.0-in. diameter, with an overall thickness of 3.6 in. It provides support for the outer storage tubes and the seating surface for the cover seal. There are twelve drilled holes in the upper support plate that accommodate the cover bolts.

The cover has a 20.0-in. diameter and an overall thickness of 1.63 in. The lower surface of the handling socket extends 8.35 in. below the bottom of the cover. Twelve holes are drilled in the cover and are sized to accommodate the closure bolts. The CCC atmosphere is free to pass between the storage tubes, but a metal Helicoflex[6] seal is provided between the container and the cover to establish container confinement. An impact limiter is located on the bottom of the handling socket closure cup. The impact limiter function is to reduce the loading to the cover that could occur during the CCC drop accident into the ISC due to the fuel assembly in the center storage location rebounding and hitting the cover. The top surface of the cover is provided with rigging attachment points for empty container handling, a test port for seal leak rate verification, and a sample port for container atmosphere sampling.

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[6] Helicoflex is a trademark of Cefilac, Societe Anonyme.

The CCC cover design provides a closure with a cover that mates to the CCC body, crushing a metallic O-ring between them, which provides confinement for the spent fuel. Leak testing requirements assure that the CCC will function as an effective "canning" barrier. After the CCC is loaded with spent fuel in the IEM cell, a pressure decay test to $\leq 1 \times 10^{-1}$ scc/sec is performed on the seal to verify correct installation of the metal Helicoflex seal. Additionally, each fuel storage tube is closed at the bottom with a nickel alloy cup to prevent potential caustic fission product solutions from degrading the ISC confinement liner in the unlikely event of a leak out of a fuel pin. Additional features of the CCC that provide "canning" assurance are: (1) all pressure boundary welds are required to be full penetration and dye penetrant inspected during the fabrication process, and (2) each CCC is hydrostatically tested to 105 lb/in^2 gauge for the design pressure of 70 lb/in^2 gauge, with a pressure of 62 lb/in^2 gauge resulting from 100% fission gas release of seven fuel assemblies.

D2.6.2.1.1 Primary Confinement.

The FFTF fuel cladding is intact and credited as the primary confinement boundary, thus meeting the intact fuel criteria when the fuel is placed into the CCC. Cladding integrity is verified when each spent fuel assembly is washed using the sodium removal process in the IEM cell. A quantitative assessment of cladding integrity is performed before transfer to dry storage; in addition, contamination of the sodium removal wash water provides an indication of a gross cladding failure during this process.

D2.6.2.1.2 Secondary Confinement.

Interim Storage Cask

The ISC is an aboveground concrete and steel shielded, top-loading spent fuel storage cask that will be used to provide safe interim dry storage of a CCC with FFTF spent fuel assemblies or pin containers for a period of up to 50 years. One CCC can be stored in the cavity of each ISC. The ISC design consists of an all-stainless steel internal confinement structure surrounded by steel and concrete shielding. The fully loaded cask weighs a maximum of 114,200 lb, including a loaded CCC with a gross payload of 5,000 lb, the closure hardware, and the weather cover. Outer cask dimensions are 85 in. in diameter and 181 in. tall. The internal cavity of the ISC is 21 in. in diameter and 147 in. tall, and will accept one CCC. This cavity, which is formed by a 1.5-in. thick stainless steel cylinder and 8-in. thick top and bottom closure plates, provides the confinement boundary.

The ISC has been designed and fabricated to meet the requirements of WHC-S-4100, in accordance with 10 CFR 72. The ISC is designed to provide confinement for the fuel, passive heat removal, and environmental protection for the CCC. It also provides radiological shielding protection for site personnel by limiting the dose rate at normally accessible surfaces to acceptable levels. A gasketed weather protection cover is installed on each ISC in the ISA. An additional cover plate may be seal welded over the bolted closure plug after receipt at the 200 Area ISA to enhance the long-term storage configuration.

The ISC confinement boundary design and analysis were performed by General Atomics, as documented in Document No. 910683, *FFTF Spent Fuel Interim Storage Cask Design*

Analysis Report (General Atomics 1995), which holds an ASME Certificate of Authorization N for design and overall fabrication responsibility for ASME Code, Section III, Division 1 and 2 components. The ISC design analysis and material properties were based on the requirements of the ASME Code, Section III, Subsection NC. Fabrication of the ISC is based on ASME Code, Section III, Article NC-4000. The ISC confinement boundary is constructed of ASTM materials. Additionally, all ISC design and fabrication activities were required to be performed using a 10 CFR 72 quality assurance program or equivalent.

The bottom of the ISC cavity is fitted with an aluminum crush pad to limit CCC impact loads in the unlikely event that it is dropped into the ISC during loading. The surfaces of the cavity are finished to remove burrs, sharp corners, and weld beads that potentially could interfere with cask loading operations. The confinement structure described above also is surrounded by annular steel shield plates that are surrounded by concrete reinforced with rebar. As discussed in General Atomics (1995), the concrete shielding structure is designed to meet ACI-349, *Code Requirements for Nuclear Safety Related Concrete Structures*. The concrete design mix was selected to ensure strength and long life over the range of temperature conditions expected during normal operations and the more extreme short-term temperatures that could occur during off-normal or accident conditions.

The ISC heat dissipation is totally passive. All required heat removal will occur by conduction, thermal radiation, and convective cooling of the outer surface. The ISC also is provided with a passive cooling system that removes heat by using an internal natural circulation airflow system. The airflow system is formed by two inlet ducts, an annular gap between the confinement boundary cylinder and the inner shield, and two outlet ducts. The inlet and outlet ducts are steel-lined penetrations through the concrete that take non-planar paths to minimize radiation streaming. These penetrations consist of two 4-in. outer diameter ducts that supply air to the bottom of a 0.75-in. wide annulus gap between the confinement boundary cylinder and the inner shielding cylinder. Natural convection circulates air up the annular space and out two similar ducts at the top of the annulus.

The ISC provides several layers of shielding to ensure worker ALARA (as low as reasonably achievable) radiation protection. This protection is based on the requirement that the 200 Area ISA is unmanned. The first layer of shielding consists of a 3.0-in. thick carbon steel clamshell around the entire length of the cavity. Another partial length, 4.0-in. thick carbon steel clamshell shield is provided for additional shielding at the cavity mid-plane where the fuel section is located. The clamshell shields are designed with studs that attach to the reinforced concrete cylinder shield. The concrete is a minimum of 21.25 in. thick for additional radial shielding. Supplemental axial shielding is provided by 4.0-in. thick plates below the bottom head and below the upper closure. A design dose rate of 2 mrem/hr at accessible surfaces for normal conditions and 200 mrem/hr at the (inaccessible) bottom head is defined in WHC-S-4100. Shielding acceptance criteria in the design specification allowed a localized dose rate of 5.0 mrem/hr to account for potential shielding imperfections and localized hot spots.

D2.6.2.2 Neutron Radiography Facility TRIGA Fuel.

A Rad-Vault is used to store NRF TRIGA casks and DOT-6M containers, which hold TRIGA SNF from the NRF in the 300 Area. Each NRF TRIGA cask can hold up to 18 elements and is nominally 38 in. tall and 16 in. in diameter. The weight of the outer vessel is approximately 1,637 lb. According to WHC-SD-TP-SARP-008, *Safety Analysis Report for Packaging NRF TRIGA Packaging*, the entire NRF TRIGA cask weighs a total of 2,013 lb when filled with 18 elements. Each DOT-6M container holding one FFCR in an inner 2R container is nominally 70 in. tall and 23 in. in diameter. The weight of the loaded DOT-6M container is approximately 640 lb. The Rad-Vault is a concrete right circular cylinder with light steel reinforcement. The empty Rad-Vault weight is 63,400 lb (43,400-lb body and 20,000-lb lid) and the maximum design weight with payload is 81,400 lb; however, the maximum expected weight is approximately 77,000 lb. With the lid installed, the Rad-Vault is 111 in. tall, with an outer diameter of 114 in.

The concrete Rad-Vault is a low-level radioactive storage unit. The Rad-Vault will be placed on compacted gravel at the ISA, and top loaded with the NRF TRIGA casks and DOT-6M containers. The Rad-Vault is equipped with a removable lid that fully exposes the available internal storage volume. The lids' mating surface to the main container is sloped to prevent rain intrusion. Opposing lifting lugs are interlaced into the steel rebar and welded wire fabric, and cast into the concrete structure. Lift capacity is sufficient to allow an empty Rad-Vault to be lifted and moved by crane to a transporter. The lid must be transported separately. The Rad-Vault is not designed to be lifted loaded or with the lid installed.

Sampling capability is provided through a gas sample port that prevents leakage in and out of the Rad-Vault. Each Rad-Vault is also equipped with a pop-up vent. No gas is expected to be vented; however, the pop-up vent will provide a vented system. There are no safety requirements for the vent functions.

Neutron Radiography Facility TRIGA Cask/DOT-6M Container Confinement Boundary

The outer container of the NRF TRIGA cask is the qualified and tested confinement barrier for storage. This container has a Helicoflex metallic seal for long-term storage integrity, an elastomeric flange seal to facilitate leak testing, and an elastomeric bore seal. The quick-disconnects used for leak testing are also fitted with metallic O-ring seals at the thread interface. The port lids covering the quick-disconnects used for helium leak testing are also provided with single metallic Helicoflex seals.

The 2R inner container of the DOT-6M system is the qualified and tested confinement barrier. This 5-in. diameter pipe vessel has a bolted flange assembly retaining an elastomeric seal and a Helicoflex metallic seal; helium leak test capability is included.

Helicoflex seals are fabricated using an inner spring covered by an inner lining and a soft outer lining. The Helicoflex seal for the NRF TRIGA and DOT-6M has a Nimonic 90 Spring[7] covered by an inner lining of Inconel[8] Alloy 600 and an outer jacket of aluminum. The materials used in the linings are application-dependent and are a function of temperature and pressure. The sealing principle of Helicoflex is based, in part, on plastic deformation of the seal's outer lining. The inner helical spring provides an elastic core under bolt preload. Each coil of the

[7] Nimonic 90 Spring is a trademark of Inco Alloys International, Inc.

[8] Inconel is a trademark of Inco Alloys International, Inc.

spring behaves independently during radial compression. The all-metal design is the reason for its long life.

The Rad-Vault provides shielding, safeguards, and weather protection, but it is not a sealed system and is not designed to provide confinement.

The NRF TRIGA cask provides multiple barriers. The inner container has an elastomeric seal on the lid and a metallic seal on the clamping eye bolt. The outer container has redundant seals provided by both an elastomeric bore seal O-ring and a Helicoflex combination metallic/elastomeric O-ring seal. Both the bore seal and the Helicoflex seal were leak tested after loading prior to initial transport and storage.

The closure devices on the NRF TRIGA cask include the closure lid and the three leak test port lids. The closure lid is attached to the cask body with twelve 0.50-in. diameter cap screws. All leak test port lids are attached to the cask lid with six 0.25-in. diameter cap screws.

The NRF TRIGA cask is provided with removable test port covers that permit access to the helium leak test components in the outer container. The container was leak tested upon initial loading according to the standards of American National Standards Institute (ANSI) N14.5, *American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment*, for a maximum leak rate of 1.0×10^{-5} scc/s (air), including the test port covers. No additional testing is anticipated or required.

The DOT-6M containers used for storage of the TRIGA FFCRs provide multiple confinement barriers. The intact cladding is a barrier. The 2R inner container provides a confinement barrier with the combination metallic and elastomeric O-ring seals.

The DOT-6M container's outer stainless steel drum can be opened to permit access to the helium leak test port on top of the 2R inner container. The 2R inner container within the DOT-6M is secured by eight 0.75-in. diameter bolts. This vessel is leak tested in accordance with ANSI N14.5 standards, for a maximum leak rate of 1.0×10^{-7} scc/s (air).

D2.6.2.2.1 Primary Confinement.

Neutron Radiography Facility TRIGA Cask

For the NRF TRIGA cask, the intact cladding is the primary barrier. In order to demonstrate fuel cladding integrity, the TRIGA pool water quality was maintained with a purification system. Impurities and minerals were removed to inhibit corrosion or filming. The purification system was carefully monitored and recorded in a weekly log book. The pool water was sampled monthly and tested by analytical chemistry to determine conductivity and pH balance and to verify that no fissionable gases were present in the water, thereby demonstrating fuel cladding integrity.

DOT-6M Container

For the DOT-6M system, the intact cladding is the primary barrier. As noted above, the TRIGA pool water quality was maintained with a purification system. Impurities and minerals were removed to inhibit corrosion or filming. The purification system was carefully monitored and recorded in a weekly log book. The pool water was sampled monthly and tested by analytical chemistry to determine conductivity and pH balance and to verify that no fissionable gases were present in the water, thereby demonstrating fuel cladding integrity.

D2.6.2.2.2 Secondary Confinement.

Neutron Radiography Facility TRIGA Cask

The inner container has an elastomeric seal on the lid and a metallic seal on the clamping eye bolt. No credit is taken for confinement provided by this barrier.

The outer container of the TRIGA cask provides confinement and has redundant seals provided by both an elastomeric bore seal O-ring and a Helicoflex combination metallic-Viton O-ring seal. Both the bore seal and the Helicoflex seal are leak tested before transport and storage. Leak test port covers have a single Helicoflex metallic seal that is also leak tested.

DOT-6M Container

The 2R inner container of the DOT-6M is a qualified and tested confinement barrier. This 5-in. pipe vessel has a bolted flange assembly retaining an elastomeric seal and a Helicoflex metallic seal; helium leak test capability is included.

The stainless steel 6M drum provides a barrier for normal conditions only. This barrier contains pressure relief capability via the penetrations covered with pressure-sensitive adhesive filament according to WHC-S-0393, *Specification for a DOT 6M/2R Metal Packaging*. This drum does not provide a safety-related confinement function, but acts as an impact absorber for the 2R container.

D2.6.2.3 Commercial Light Water Reactor Fuel.

The commercial LWR fuel storage system consists of six storage units, each comprised of a welded canister, a NAC-1 cask (Figures D2-10 and D2-11), and an ISO shipping/storage container (Figures D2-12 and D2-13). Each canister will contain a single PWR fuel assembly, or in the case of the consolidated loose rods, a consolidated rod container designed to maintain the positioning upon which the criticality safety analysis is based. The welded canister provides the primary containment. Analysis shows that damage to the canister is not credible. The canister will be placed within the NAC-1 cask, which will provide shielding and structural protection of the fuel during storage.

The NAC-1 casks, previously licensed by the NRC to transport LWR spent fuel and waste material, are modified for use at Hanford for both transportation and storage purposes. The NRC license has not been retained. Modifications to the cask included the removal or plugging of several valves connected to the confinement cavity. The NAC-1 casks are mounted to supports within the ISO container for transportation and will remain in the containers in this configuration during storage at the 200 Area ISA.

The storage units will be placed by crane alongside each other (1 x 6 array) and evenly spaced 4 ft apart. This spacing is not required for criticality or other safety analysis purposes, but rather for personnel access considerations and maintaining personnel radiation doses ALARA, since this is an unmanned facility.

Light Water Reactor Canister Description

The canisters are based on criteria provided in SNF-4894, *Spent Nuclear Fuel Project Acceptance Criteria for LWR Spent Fuel Storage System*. These criteria include the specification that the design and fabrication meet the requirements of ASME Code, Section III, where necessary to maintain geometry control such that 0.95 k_{eff} is not exceeded. The design life expectancy of the canisters is 75 years.

The welded canister provides the primary confinement boundary for the fuel. The canister is fabricated from a 12-in. schedule 40 stainless steel pipe, with a welded base cap and top closure. It has a maximum outside diameter of 13.375 in., including the end cap, and a total length of 177.36 in. The bottom cap is machined, so the pipe-to-cap weld is inspectable at the side of the canister. The top closure lid weld will be tested for leak tightness per the requirements of ANSI N14.5. The closure lid contains a penetration that allows the canister to be evacuated for vacuum drying of the fuel assembles, filled with helium, leak tested, and welded closed. The canister shield plug allows the installation of handling attachments for cask loading and subsequent repackaging operations. Figure D2-14 provides an illustration of the canister.

The canisters are provided with safety-class internal supports for the fuel assemblies or the loose rod consolidation container, which provide geometry control such that a loaded canister will have a k_{eff} less than 0.95 when fully moderated and reflected under all credible accident conditions.

The maximum loaded weight of the canister is not to exceed 3,300 lb. The maximum internal design pressure of the canister is 75 lb/in^2 gauge, tested to a pressure of 100 lb/in^2 gauge during fabrication in accordance with ASME B&PV Code, Section III, Division 1, Subsection NB. The final end closure weld made after the canister is loaded with spent fuel is not required to be pressure tested, as identified in ASME B&PV Code Case N-595-1.

The canister for the consolidated loose rods requires the rods be provided with safety-class containment within the canister so that criticality limits are not exceeded. The k_{eff} of the optimum pitch of the "bundle" of fuel rods in a canister is below 0.95 when the fuel is placed within the rod consolidation container, even if the canister is flooded. In addition, as discussed in Section D6.4.1.2, it is considered incredible to flood the horizontal welded canister at the 200 Area ISA. Figures D2-14 and D2-15 depict the consolidated rod container. The consolidated rod container design provides for helium flow to be in contact with the fuel rods for vacuum drying and helium backfill.

Cask Description

The acceptance criteria require that the NAC-1 cask is designed and fabricated to the requirements of the ASME Code, Section III (1971). The external shape of the NAC-1 spent fuel shipping cask approximates a smooth-surface, right circular cylinder that is modified, in that impact limiters protrude radially at both ends. The internal cross-section of the cask cavity is circular. The overall dimensions of the cask include a length of 214 in. (including lid impact limiter) and a maximum cross-sectional envelope diameter of 50 in. The internal cavity of the cask is 178 in. long and 13.5 in. in diameter. The maximum loaded gross weight of the cask, including the maximum fuel and canister weight (3,300 lb), is approximately 49,000 lb. The principle design features of the cask are the transportation confinement boundary, shielding and heat dissipation systems, and the lifting and tie-down systems.

The structures of the NAC-1 cask are constructed of stainless steel. The cask cavity is formed by the inner shell, which is a 14.125-in. outside diameter, 0.3125-in. thick stainless steel shell. The upper end of the shell is welded to the cask cavity flange; the bottom end of the shell is welded to the cask bottom casting. Surrounding the inner shell of the cask is a nominal 6.6875-in. thick annulus of chemical grade lead (gamma) shielding. The lead is shaped such that approximately 5 in. from the bottom and 30 in. from the top, the thickness is reduced to 5.4375 in. There is an annular void, 5 in. long by 1.25 in. thick, at the bottom end of the gamma shield to allow for any lead expansion during the fire accident. The upper axial shaping is accomplished by reducing the diameter of the outer shell 2.50 in. over a 30-in. length. The lead/steel interface of the inner and outer shell have axial copper fins that are imbedded in the lead and welded to the inner and outer shells to transfer heat across the interface with a minimum temperature gradient.

The confinement boundary of the NAC-1 cask is the inner shell, lower end casting, upper end casting, bolted closure lid with double polytetrafluoroethelene (PTFE) O-ring seals and seal test port, a helium fill/vent valve, rupture disks, and lower casting drain valves. Because the welded canister provides the final confinement boundary for storage, it will not be required to inspect and replace the O-ring seals. The NAC-1 cask will be modified to accept the designed canister and further minimize worker exposure by eliminating unnecessary surveillance and maintenance activities during storage at the 200 Area ISA. The modifications, in accordance with SNF-4894, consist of removing the anti-rotational lugs in the cask cavity and replacing drain, relief, valve penetrations, and rupture disks with pipe plugs. The cover plates will be reinstalled over the ports, as designed, to serve as heat shields during the postulated design basis fire accident. Also, the neutron shield tank burst disk assembly port penetrations will be plugged and have the muffler removed.

The outer shell is formed by a 30-in. diameter, 1.25-in. thick stainless steel cylinder reduced to a 27.50-in. diameter at one end. The cask bottom consists of a shaped stainless steel disc with a 30-in. outer dimension, and an 8-in. thickness that functions as a gamma shield for the bottom end of the cask. The cavity flange is a stainless steel ring with a 29.75-in. outer dimension, 17.50-in. inner dimension, and an 8.625-in. thickness. The bottom disc end and top flange are welded to the inner and outer shells to form the enclosure for the lead gamma shield.

Neutron shielding tanks are provided in the 4.50-in. thick annular space formed between the outer shell of the lead gamma shield and a thin stainless steel shell that constitutes the outer cask surface. The neutron shield tanks will not be used for the storage configuration at the 200 Area ISA and will contain air.

Upper end shielding is provided by the 7.50-in. thick stainless steel cask lid. The cask lid is a stainless steel casting that also serves as a gamma shield. The lid is a flanged frustum of a cone, 7.50 in. thick, with a maximum diameter of 25.50 in. The conical portion of the stainless lid is stepped locally to minimize the gap between the lid and cavity flange, preventing displacement of the lid during the design basis drop. The flanged portion of the lid is a 2-in. thick, 25.50-in. diameter disc. There are six counter-bored clearance holes for the lid bolts and four 1-in. diameter blind-threaded holes for attaching the lid impact limiter. The cask lid is bolted to the cavity flange by six 1.25-in. diameter hex-head bolts. Bolt heads bear on the cask lid; the shanks penetrate through the lid flange and thread into the cavity flange. Six 1.25-in. diameter holes with HeliCoil[9] thread inserts are provided in the cavity flange for bolting the cask lid to the flange. The bolt heads are drilled for a wire security seal.

The lower end impact limiter structure is a ring that surrounds the cask lower casting, formed from a stainless steel sheet and/or plate welded to the cask outer shell and flange areas. The impact limiter was designed to absorb the energy of the design basis 30-ft free drop. It contains a balsa wood disc placed adjacent to the cask bottom. A 0.125-in. thick sheet of asbestos is placed between the balsa and the cask bottom, and is contained internally. Within the impact limiter, extending radially from the center of the cask, are eight 0.375-in. thick stainless steel gussets. The bottom section of the impact limiter also functions as a pedestal for supporting the cask in the vertical position.

An impact limiter is located at the upper end of the cask body, designed to absorb the energy of the design basis side drop accident. The upper impact limiter is a stainless steel-sheathed, balsa-filled ring that surrounds the cask cavity flange. A 0.125-in. sheet of asbestos is positioned between the balsa and cask outer shell, and is contained internally. Within the impact limiter, extending radially from the center of the cask, are eight 0.375-in. thick stainless steel gussets. The upper impact limiter is also a cask support member in the storage configuration, resting in the ISO cradle frame cross-member.

The cask lid is protected during impact by a 12-in. thick, balsa-filled lid impact limiter that covers and overlaps the cask lid and cavity flange. The balsa is enclosed within a 0.109-in. thick stainless steel container. A 0.125-in. sheet of asbestos is positioned between the balsa and the sheet material adjacent to the cask, and is contained internally. The impact limiter is attached to the cask lid by four 1-in. diameter bolts. There are elastomer O-rings in grooves under the heads of the 1-in. bolts and a neoprene gasket on the perimeter of the impact limiter. These impact limiter seals are for weather protection and do not provide a confinement function. Radiation exposure to the seals is not a failure concern. Removal of the lid impact limiter allows access to the cask lid.

Lifting devices for the NAC-1 cask are designated as lifting trunnions and rotation trunnions. The lifting trunnions are two 8.625-in. diameter by 3-in. long trunnions located on the perimeter of the upper impact limiter. The cask is lifted by a special handling yoke attached to the two trunnions. The rotation trunnions are two 6.625-in. diameter by 3-in. long trunnions for rotating the cask to and from the horizontal position in the ISO container. The lower trunnions are offset from the cask centerline so that when the cask is lowered into the ISO container, it will rotate to a horizontal position as the crane hook is lowered.

Transportation of the NAC-1 cask is in the normal storage configuration with the cask in a horizontal position, secured within the ISO container. Two structural cross-member sections

[9] HeliCoil is a trademark of Cefilac, Societe Anonyme.

serve as cradles for the cask within the ISO container. The lower (rotation) trunnions of the cask are captured by a notch and clamping plate on the aft cradle. The upper lifting trunnions are also captured by clamping plates, to hold the upper end of the cask as it rests on the impact limiter within a neoprene-lined forward cradle. The lid impact limiter is bolted to the cask lid. The lid impact limiter can be unbolted and rolled away from the cask on a track. The cask and ISO container are lifted by crane from the transport trailer and placed in the storage array on the assigned concrete pad.

International Standards Organization Container Description

The NAC-1 cask is transported and stored within a specially designed ISO shipping container. These containers are similar in design and appearance to SEALAND containers. The ISO container provides weather protection for the cask, which is not a safety function. The ISO container is designated safety-significant for structural integrity. The ISO containers were fabricated in two heights, 6-ft and 8-ft high models. All containers are painted carbon steel construction. Figure D2-12 shows the basic details of the ISO container. The safety-significant components of the ISO container include the trunnion support structures and the structural members, as depicted in Figure D2-13.

The 8-ft ISO container, fabricated by Evergreen Heavy Industrial Corporation, is nominally 8 ft high x 8 ft wide x 20 ft long and has an empty weight of 8,624 lb. The container has single full-width doors at each end and a removable roof. The frame of the container is structural steel channel construction, with a 0.50-in. carbon steel floor. The sides are fabricated of 0.063- and 0.079-in. corrugated carbon steel. The roof is fabricated of 0.063-in. carbon steel sheet metal supported by 1.575-in. x 1.575-in. angle iron on a 23.622-in. grid, and is slightly pitched to prevent ponding of precipitation. The doors and roof are provided with weather seals. Two of these containers will be used to house NAC-1 casks during fuel transport and storage.

The 6-ft ISO container, fabricated by Adamson Containers Ltd., is 6 ft high x 8 ft wide x 20 ft long and has an empty weight of 4,400 lb. The container has two half-width doors at one end and a removable roof. Materials of construction and dimensions are similar to those used in fabricating the 8-ft containers, except the roof material is 0.055-in. carbon steel sheet metal and the roof is flat. Four of these containers will be used to house NAC-1 casks during fuel transport and storage.

D2.6.2.3.1 Primary Confinement.

Although past inspection of the fuel assemblies and rods has shown no visible damage, cracks, or pin holes, and additional visual observations will be performed during loading activities, analyses have assumed damaged fuel to be bounding if any cladding failure is identified. The verification process meets the requirements of 10 CFR 72 for cladding inspection to assess the integrity of the cladding. Since damaged fuel is assumed in the analyses, the LWR canister is credited to provide primary confinement. The canister has a double welded closure, is leak tight, and analysis demonstrates that none of the accident conditions result in breach of the canister. This configuration provides two confinement boundary welds and is considered to meet the requirement of assured confinement. The welded canister also satisfies the fuel retrieval requirement of 10 CFR 72. The inner canisters are backfilled with helium and relieved to atmospheric pressure prior to final seal welding. The presence of helium is not required for heat transfer purposes, nor to maintain fuel integrity. Each spent fuel canister will be contained within a NAC-1 cask for shipment and storage at the Hanford Site.

D2.6.2.3.2 Secondary Confinement.

With double welded canisters providing the primary confinement, a secondary confinement boundary is not required and the NAC-1 storage cask is not relied on for confinement.

D2.7 SAFETY SUPPORT SYSTEMS

This section identifies and describes the principal systems that perform safety support functions.

D2.7.1 Criticality Prevention

D2.7.1.1 Fast Flux Test Facility Fuel.

A criticality safety evaluation was performed for storage of the FFTF SNF in the 400 Area ISA and provides the bases for satisfying the criticality prevention criteria of double contingency to K_{eff} less than 0.95. The criticality safety evaluation is documented in WHC-SD-FF-CSER-004, *Criticality Safety Evaluation for Long Term Storage of FFTF Fuel in Interim Storage Casks.* Since the configuration will not change for the 200 Area ISA, the calculations and evaluations performed for the 400 Area ISA are valid. In particular, the CCC was analyzed with respect to its structural and confinement effectiveness as a canning structure for FFTF fuel. The results for the CCC demonstrate that the CCC retains its confinement and structural integrity under all normal and accident conditions where all loads were found to be within the Code Class limits, g allowables, and stress allowables for the CCC.

D2.7.1.2 Neutron Radiography Facility TRIGA Fuel.

There are no components performing a criticality prevention function requiring safety classification (Chapter D6.0). The criticality safety evaluation was performed for the storage of NRF TRIGA SNF in the 400 Area ISA and provides the bases for satisfying the criticality prevention criteria, including double contingency and K_{eff} less than 0.95. The TRIGA fuel configuration at the 200 Area ISA is identical to the 400 Area configuration.

D2.7.1.3 Commercial Light Water Reactor Fuel.

Separate criticality safety evaluation reports were prepared to address interim storage of PWR fuel assemblies and loose fuel rods including BWR fuel in the NAC-1 casks in the ISA. These reports and Chapter D6.0 address specific criticality aspects of LWR fuel storage in the ISA as individual casks and in the storage array. The LWR canister is required for criticality geometry control and is designated safety class.

The canisters for the PWR fuel assemblies, and the canister internals, will provide the safety-class support to retain the fuel material within the square cross-sectional geometry of the fuel assembly. The canister is also leaktight, precluding water intrusion, which also makes an accidental criticality incredible. For the BWR and PWR loose rods, the consolidated rod container along with the canister and internal support basket will provide safety-class geometry control for the bundle of loose rods.

D2.7.2 Fire Protection

The ISA is constructed of non-combustible materials and there are no fire hazards inherent to the ISA (SNF-4932). Administrative controls ensure that the area within the fence is kept free of debris and vegetation. The fence, provided for access control and radiological exposure control purposes, will also keep transient debris from entering the ISA. The only credible fire hazard associated with the ISA is the equipment used to transfer and handle the storage systems. Accidents associated with the ISA have been determined to be bounded by the transportation fire scenario of Title 10, *Code of Federal Regulations*, Part 71, "Packaging and Transportation of Radioactive Material" (10 CFR 71), Section 71.73(c)(3). Specific accidents that are bounded include the transporter tractor vehicle, mobile crane fire, and runaway fuel truck. Therefore, the maximum credible fire for the ISA is bounded by the transportation fire scenario.

D2.7.2.1 Fast Flux Test Facility Fuel.

The ISC is constructed of nonflammable materials. The design basis storage fire for the ISC is bounded by the design basis transportation fire defined in 10 CFR 71, as discussed in the 200 Area ISA fire hazards analysis (SNF-4932).

D2.7.2.2 Neutron Radiography Facility TRIGA Fuel.

The TRIGA cask is constructed of nonflammable materials. The survival of the TRIGA cask is demonstrated by the analysis of the design basis fire. Since the NRF TRIGA cask is removed from the impact limiters that were analyzed as part of the transportation fire in WHC-SD-TP-SARP-008, a supplemental fire analysis was performed for the storage configuration inside the Rad-Vault.

The DOT-6M container is constructed of nonflammable and fire retardant materials.

D2.7.2.3 Commercial Light Water Reactor Fuel.

The NAC-1 and the ISO are constructed of nonflammable materials. The survival of the NAC-1 is demonstrated by the analysis of the design basis fire.

D2.7.3 Cask Instrumentation

Continuous monitoring of parameters of the dry storage systems (e.g., interseal pressure and temperature) are not provided. While such monitoring is recommended for mechanical closures by NUREG-1536, page 7-3, the FFTF and TRIGA fuel have demonstrated intact fuel cladding prior to placement into storage, and the storage system confinement boundaries are tested to a leak tight condition, where leaktight is as defined by ANSI N14.5. The FFTF and TRIGA fuels are protected by multiple confinement barriers and have smaller releasable gas and volatile radionuclide source terms compared to commercial LWR fuel dry storage systems. The LWR fuel stored in the NAC-1 casks is confined in a seal-welded container, which is also tested to leaktight conditions. Large margins in the thermal design of the storage systems preclude the need for temperature monitoring to ensure fuel integrity. The dry storage systems are described below.

Surveillance of the ISA is required to assure safe storage (NUREG-1536, page 7-3), when cask parameter monitoring is not used. This surveillance assists in the early identification of conditions that could lead to unsafe storage. The surveillance requirement is met by a combination of closed circuit television monitoring or frequent (daily) walk through (or walk by) inspections for fence security, fire watch, roving patrol, or other periodic checks. Surveillance procedures will be implemented prior to ISA operation. Closed circuit television monitoring (if used) will be in a continuously manned location (i.e., CSB control room or patrol headquarters).

D2.7.3.1 Fast Flux Test Facility Fuel.

Only intact FFTF fuel is contained within the FFTF ISC, which is a stainless steel and concrete cask, with the confinement boundary provided by the stainless steel shell, closure lid, and double metallic O-ring seals. The confinement boundary is designed to the ASME Code, Section III, Subsection NC (1989). The double metallic O-ring seals are tested to leaktight conditions, and then the penetration port between the O-ring seals is closed by welding the penetration port covers in place. An additional cover plate may be seal welded over the bolted closure plug after receipt at the 200 Area ISA to enhance the long-term storage configuration.

Because none of the design basis accidents can result in a breach of the ISC confinement boundary, and normal storage environmental conditions are well within the design parameters of the metallic Helicoflex double seals, continuous pressure monitoring is not provided for the interseal space.

The FFTF ISC uses air channels to provide for cooling air flow along the walls of the vessel that provide confinement. Historically, the NRC has required the monitoring of outlet temperatures of casks to demonstrate that the thermal performance of the cask is not degrading. Thermal analysis (WHC-S-4100) has determined that the cask thermal performance is acceptable with the absence of cooling air flow. Thus, no loss of confinement events occur due to high temperature. No monitoring of the outlet air temperature of the FFTF ISC is required or provided.

The FFTF ISC design incorporates provisions for continuous monitoring of cask parameters, if needed for future use. Conduit runs for wiring have also been provided in the FFTF storage pad of the 200 Area ISA.

D2.7.3.2 Neutron Radiography Facility TRIGA Fuel.

Neutron Radiography Facility TRIGA Cask

TRIGA fuel that is considered to be intact is loaded into NRF TRIGA casks. The TRIGA casks are closed after loading and leak tested to leaktight conditions by testing between a metallic closure seal and a Viton O-ring.

This design does not meet the requirement for double metallic O-rings (NUREG-1536, page 7-2), but this fuel has a very small source term for gas and volatile radionuclides. Since the potential releasable source term is small and there are no design basis accident mechanisms for seal failure, monitoring of the confinement boundary does not seem justified and is not provided.

The TRIGA cask incorporates passive heat rejection from the body of the cask and does not require thermal performance monitoring.

The TRIGA cask design incorporates provisions for continuous monitoring of cask parameters, if needed for future use. Conduit runs and wiring can also be provided to the Rad-Vault gravel area, if necessary.

DOT-6M Container

TRIGA FFCRs are loaded into a DOT specification 2R container. The 2R container incorporates a leak test port and double O-rings (one metallic and one elastomer). When closed, the 2R is tested to demonstrate a leak tight seal.

The design of the 2R container incorporates a metal seal in the outer groove of the container and an elastomer Viton O-ring in the inner groove. If the Viton O-ring fails, the interseal test port becomes part of the confinement boundary. To preclude possible leakage from the test port, the test port plug is also fitted with a metallic gasket. Radiation exposure to the seals is not a failure concern.

Since the potential releasable source term is very small and there are no design basis accident mechanisms for seal failure, monitoring of the confinement boundary does not seem justified and is not provided.

The 6M drum incorporates passive heat rejection from the body of the drum and does not require thermal performance monitoring.

There are no design features that can be used for parameter monitoring of the 2R/6M drum configuration. Consequently, parameter monitoring cannot be implemented for this configuration.

D2.7.3.3 Commercial Light Water Reactor Fuel.

The LWR fuel is confined in the LWR canister, which is closed by double welding. After the first weld of the shield plug, the canister is tested to leaktight conditions, as defined by ANSI N14.5, and a second closure plate is welded in place. This configuration provides two confinement boundaries and is considered to meet the requirement of assured confinement. In accordance with NUREG-1536 (page 7-4), no confinement monitoring will be required. After loading the LWR canister into the cask, the cask will be closed using the containment boundary PTFE O-rings, but these O-rings are not required to maintain integrity for extended storage.

The NAC-1 cask incorporates passive heat rejection from the body of the cask and does not require thermal performance monitoring. The decay heat of the stored LWR fuel is significantly less than the design basis heat load of the cask. According to NAC-E-804, *Safety*

Analysis Report for the NFS-4/NAC-1, Spent-Fuel Shipping Cask (NAC 1990), the NAC-1 cask is designed for heat loads up to 750 W.

Since the LWR canister is a welded closure, no confinement monitoring is required. The NAC-1 cask and LWR canister designs do not incorporate provisions for continuous monitoring of any cask parameters.

D2.7.4 Radiation Monitoring

10 CFR 72, Sections 122(I) and 126(c)(2), require the ability to monitor radiation levels. At the ISA, since the radiation doses are not anticipated to change after cask placement, the most appropriate means of doing this is dosimetry, alarming supplemental dosimetry, and periodic measurements by radiological control technicians. An area radiation monitor is not provided. Thermoluminescent dosimeter monitoring on the 200 ISA perimeter fence will demonstrate compliance with radiation dose limits imposed by 10 CFR 72 and monitored in accordance with 10 CFR 835.

D2.7.5 Cranes

The cranes to be used include the following:

- 150-ton Manitowoc 4000W crawler
- 250-ton Manitowoc 4100 crawler
- Hydro-Crane for TRIGA.

The above cranes have been analyzed for the crane fall accident in Section D3.4.2. A Technical Safety Requirement precludes use of other cranes in the ISA unless analyzed and approved.

D2.7.6 Lifting Equipment

All hoisting and rigging evolutions will be conducted in accordance with DOE/RL-92-36. All lifts of SNF or casks containing SNF will be critical lifts.

D2.7.6.1 Fast Flux Test Facility Fuel.

The lifting equipment interfaces with the three removable ISC lifting lugs attached to the crane (Figure D2-2). The three ISC lifting lugs are equally spaced on a 62.0-in. diameter circle on the top of the ISC. The spacing aligns directly with the existing spreader assembly (drawing H4-65153-A1). Three existing tension-ties (drawing H4-65153-A2) are used to connect the spreader assembly to the 1.75-in. thick lifting lugs. The fully loaded ISC weighing 114,200 lb uses 86% of the safe working load of the three tension-tie assemblies. Lifting will be covered by an approved procedure.

D2.7.6.2 Neutron Radiography Facility TRIGA Fuel.

Lifting of the TRIGA cask is performed with two swivel hoist rings that thread into the cask cover. These rings are connected with two slings to a lifting beam, which connects to the crane hook with single slings and a load cell. The loaded TRIGA cask weighs approximately 2,000 lb. Lifting will be covered by an approved procedure.

The DOT-6M is lifted with a commercial drum lifter rated at 1,000 lb capacity, which connects to the crane hook using single slings and a load cell. The DOT-6M weighs approximately 650 lb. Lifting will be covered by an approved procedure.

The Rad-Vault lid is lifted with four slings connected to a lifting beam, which then has two slings to the crane hook with a minimum sling angle of 60 degrees. The Rad-Vault lid weighs approximately 20,000 lb. Lifting will be covered by an approved procedure.

D2.7.6.3 Commercial Light Water Reactor Fuel.

The lifting equipment interfaces with the four ISO corner lifting lugs, the ISO lifting fixture, and with the 250-ton crane. The fully loaded ISO weighs a maximum of 56,000 lb. Lifting will be covered by an approved procedure.

D2.8 UTILITY DISTRIBUTION SYSTEMS

D2.8.1 Interim Storage Area Power Supply System

The ISA has power at the fence for the sodium lighting system. The lighting provides a security function for the required drive-by inspections.

The conduit embedded in the concrete pads is PVC GRC meeting the NEMA Standard RN-1. The conduits are terminated with a rigid steel coupling and pipe plug, installed flush with the top of the concrete. Conduit ends terminating in vaults or handholds are terminated with ground bushings. Ground bushings are connected in series with a minimum of #12 AWG bare copper wire to six ground conduits of 0.75 in. diameter.

The ISC pads have conduits for power and signals. There are five handholds located along the outer longer edges of each ISC pad; the first being ~2.5 ft from the north edge, and the next four being spaced at approximately 30 ft to each successive one. After the fourth handhold, to the south, the PVC GRC drops from a 4-in. to 2-in. diameter, and is a single run to the fifth handhold. Signal conduit runs from the handhold out under the concrete pad and terminates to the six successive storage locations. This conduit is buried at a depth of 24 in. below grade. The power and signal conduits become coincidental at 24 ft from the northern-most outer edge of each ISC concrete pad.

The NAC-1/ISO pad has conduits for power and signals. The single handhold for the NAC-1/ISO pad is centrally located along the northern 28-ft edge, 2.50 ft from the pad. Seven conduits of 0.75-in. PVC GRC run from the handhold, and one of the seven is terminated at each of the seven proposed storage locations. They are located 1-ft north and 3-ft west of the northeast corner of each storage container. This conduit is buried at a depth of 24 in. below grade. The power and signal conduits become coincidental, as the pad is centrally located along the northern 28-ft edge at 2.50 ft from the pad, 35 ft from the northern-most outer edge of each NAC-1/ISO concrete pad.

D2.8.2 Interim Storage Area Piped Utility System

One fire hydrant (1,000 gal/min) is located approximately 152 ft from the southwest corner of NAC-1/ISO pad, just outside the fence line (Drawing H-2-829294). There is also an 8-in. fire main connection at the proposed ISA storage building.

D2.8.3 Interim Storage Area Communications System

Currently, no monitoring system is identified for use at the ISA. Conduit is provided for flexibility and future use. Communication will be provided via portable handheld equipment.

The conduit embedded in the concrete pads is PVC GRC meeting the NEMA Standard RN-1. The conduits are terminated with a rigid steel coupling and pipe plug, installed flush with the top of the concrete. Conduit ends terminating in vaults or handholds are terminated with ground bushings. Ground bushings are connected in series with a minimum of #12 AWG bare copper wire to six ground conduits of a 0.75-in. diameter.

The ISC pads have 4-in. and 2-in. conduits for signal wires. There are five handholds located along the outer longer edges of each ISC pad. The first being ~2.5 ft from the north edge, and the next four being spaced at approximately 30 ft to each successive one. After the fourth handhold, to the south, the PVC GRC drops from a 4-in. to 2-in. diameter and is a single run to the fifth handhold. Signal conduit runs from the handhold out under the concrete pad and terminates to the six successive storage locations. This conduit is buried at a depth of 24 in. below grade. The power and signal conduits become coincidental at 24 ft from the northern-most outer edge of each ISC concrete pad.

The NAC-1/ISO pad has 2-in. conduits for signal wires. The single handhold for the NAC-1/ISO pad is centrally located along the northern 28-ft edge, 2.50 ft from the pad. Seven conduits of 0.75-in. PVC GRC run from the handhold, and one of the seven is terminated at each of the seven proposed storage locations. They are located 1-ft north and 3-ft west of the northeast corner of each storage container. This conduit is buried at a depth of 24 in. below grade. The power and signal conduits become coincidental at 35 ft from the northern most outer edge of each NAC-1/ISO concrete pad.

D2.9 AUXILIARY SYSTEMS AND SUPPORT FACILITIES

D2.9.1 Lightning Protection

D2.9.1.1 Fast Flux Test Facility Fuel.

Results of a lightning analysis, documented in SNF-4791, *200 Area Interim Storage Area Technical Information Documentation for FFTF Fuel Storage*, demonstrate that lightning strikes will not credibly lead to the release of radioactive materials from the FFTF ISC. The analysis shows that effects of a lightning strike to the ISC cask are minimal. A direct strike on the cask would not produce significant temperature rise or damage to the cask contents. Subsequent to a likely voltage breakdown in the top concrete layer during a direct, worst-case 218,000 amp lightning strike to the cask, current flow during the discharge would pass along the concrete reinforcing rod structure and from there to ground. Current flow through the internal cask materials inside of the inner shield would be negligible due to the electrical skin effect. Lightning cannot lead to the release of radioactive material; therefore, no lightning protection is required.

D2.9.1.2 Neutron Radiography Facility TRIGA Fuel.

A lightning analysis was performed that evaluated the effects of a lightning strike on a Chem-Nuclear Systems, Inc. Type 14-215 Rad-Vault storage container. The container consists of a concrete structure with steel reinforcement. The analysis, which is documented in Section 3.0 of SNF-4793, *200 Area Interim Storage Area Technical Information Documentation for TRIGA® Fuel*, assumed that the cask was struck with the full energy of the maximum credible lightning strike. The analysis shows that effects of a lightning strike to the Rad-Vault cask are minimal. A direct strike on the cask would not produce a significant temperature rise or damage to the cask contents. Subsequent to likely voltage breakdown in the top concrete layer and during the event of a direct worst-case 218,000 amp lightning strike to the cask, current flow during the discharge would pass along the concrete reinforcing rod structure and from there to ground. Current flow through the internal regions of the Rad-Vault wall would be negligible due to the electrical skin effect. Spalling of concrete at both the top and bottom areas of the cask could occur at the regions where the current penetrates the concrete and enters or exits the steel reinforcing wires. Arc discharge would likely occur in the gasket area, possibly exposing interior gases to an ignition source. No flammable gases are anticipated inside the vented Rad-Vault containing leaktight containers. Lightning cannot lead to the release of radioactive material; therefore, no lightning protection is required.

D2.9.1.3 Commercial Light Water Reactor Fuel.

Analysis documented in SNF-4795, *200 Area Interim Storage Area Technical Information Documentation for LWR Fuel Storage*, evaluated damage resulting from a lightning strike to the cask/container system. The analysis concludes that even in the most severe case where the cask is not protected by the surrounding ISO shipping container, a direct lightning strike would not produce significant temperature rise or damage to the cask surface materials, fuel contents, or confinement function.

Current flow during the discharge of a lightning strike would pass along the stainless steel surface of the cask and from there to ground. Adequate electrical bonding between the cask and storage container is provided by the cask rotation trunnion clamps. Current flow through the interior of the cask would be much less than the surface due to the electrical skin effect. In the event that a maximum current level strike occurred on the cask top end, current flow would spread along the surface on its path to ground. Maximum temperature rise at the surface of the cask during the lightning strike is estimated to be 28 °F.

The lightning analysis does not take credit for the cask being stored in the ISO shipping container. The ISO shipping container provides additional lightning protection for the cask, as it will dissipate current in a similar manner as the cask surface during a lightning strike, preventing any flashover of electrical current to the cask. No lightning protection is required.

D2.10 REFERENCES

10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*.

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

10 CFR 73, "Physical Protection of Plants and Materials," *Code of Federal Regulation*.

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10 CFR 835, "Occupational Radiation Protection," *Code of Federal Regulations*, as amended.

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Figure D2-1. 200 Area Interim Storage Area.

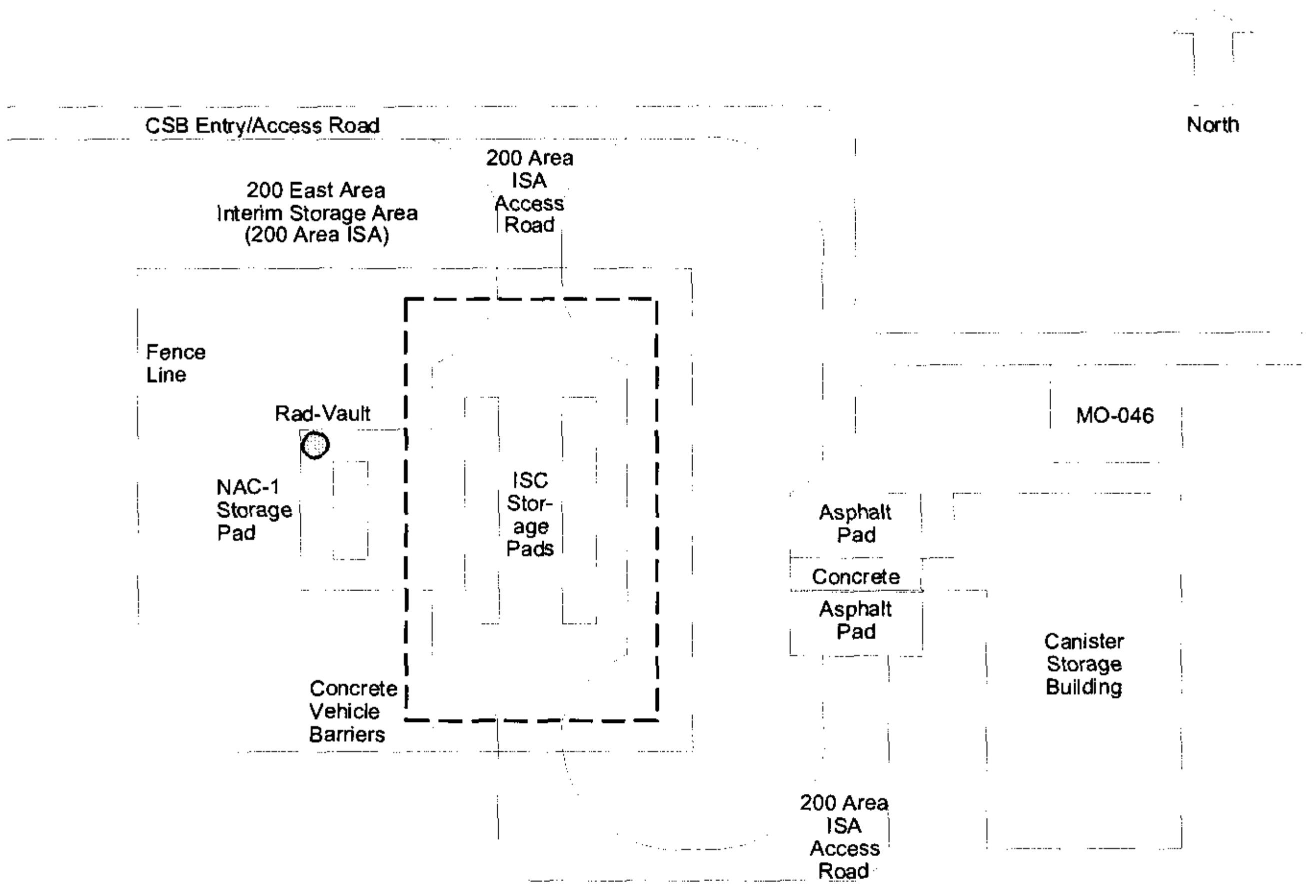


Figure D2-2. Interim Storage Cask with Lift Fixture Attached to Mobile Crane

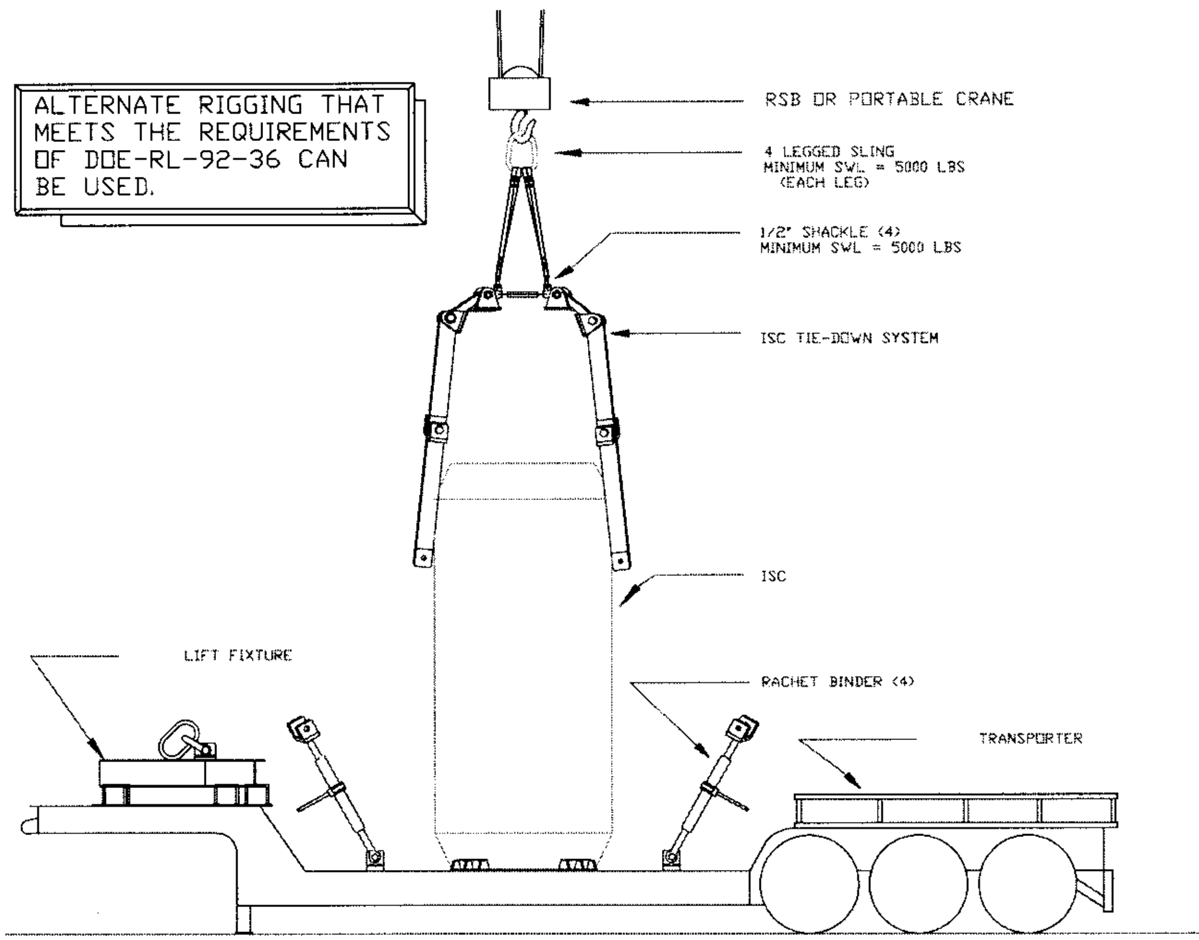


Figure D2-3. Rad-Vault.

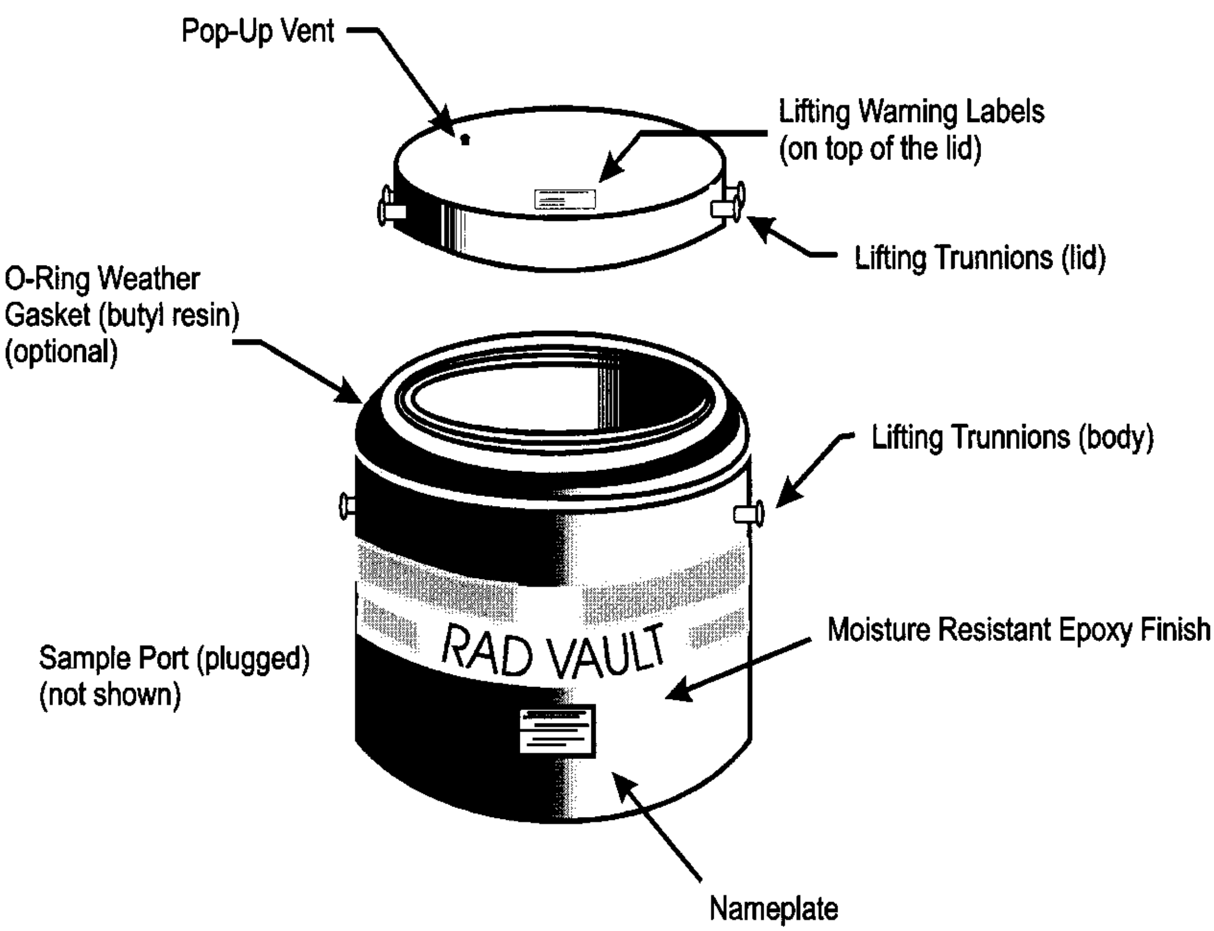


Figure D2-4. NRF TRIGA Cask - Arrangement.

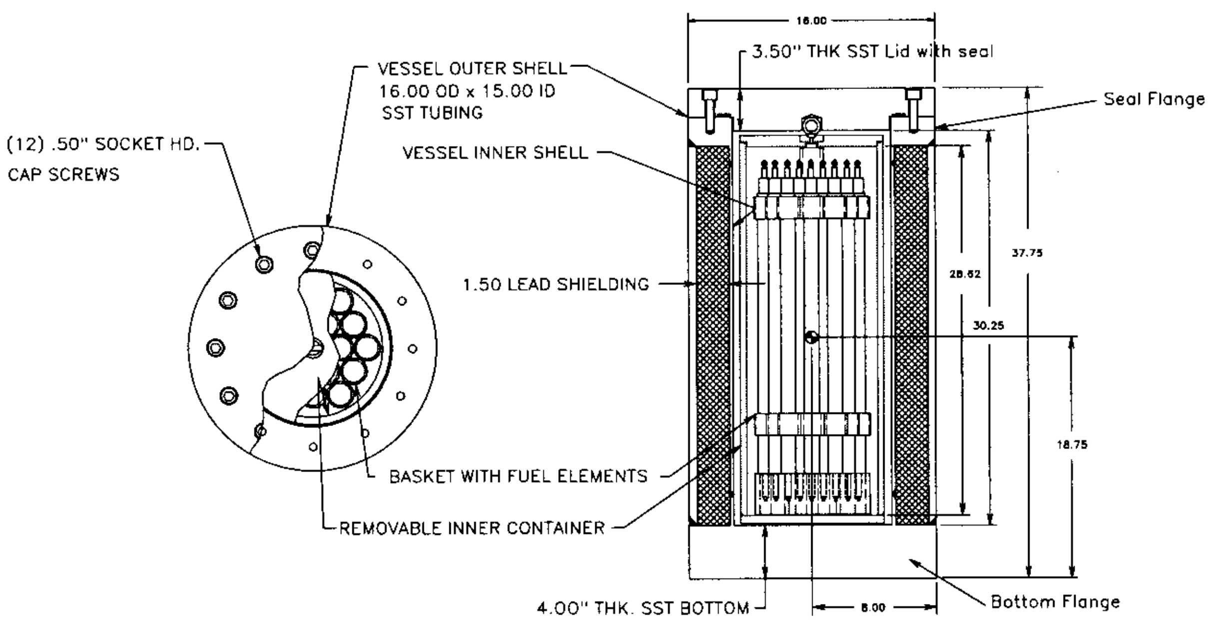


Figure D2-5. NRF TRIGA Cask - Outer Vessel.

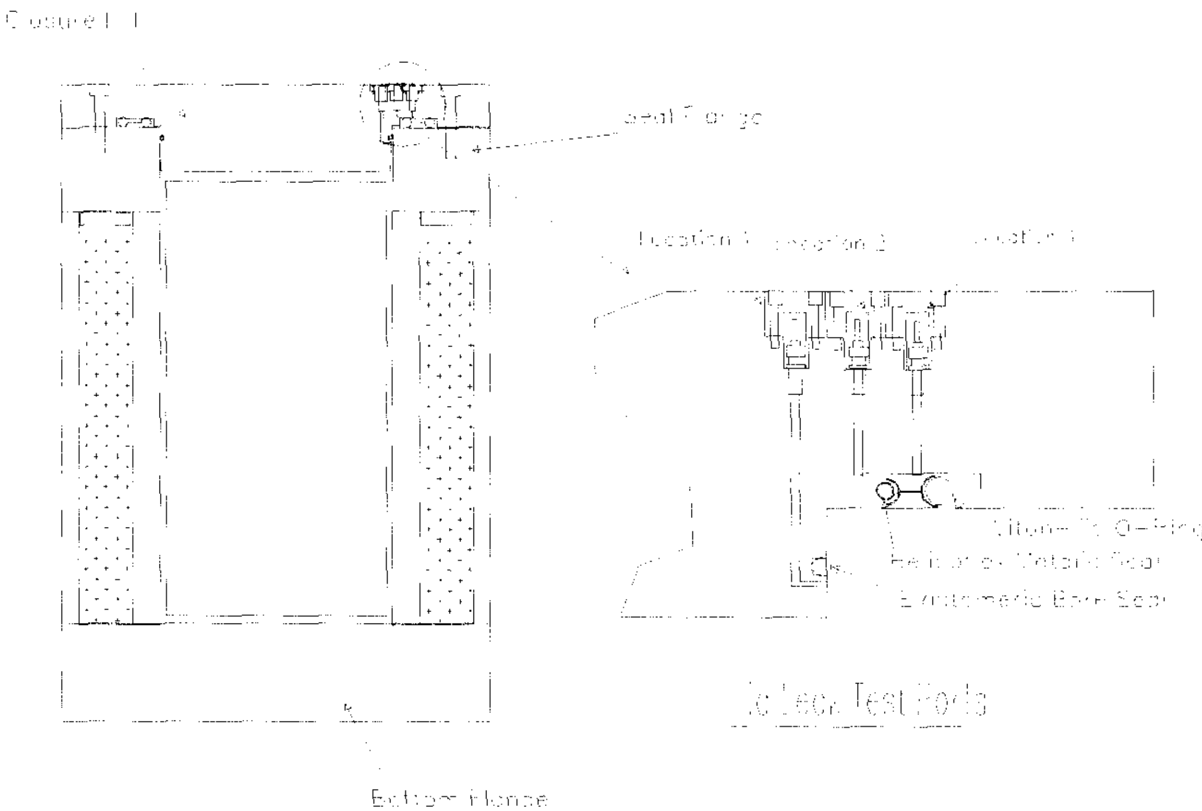
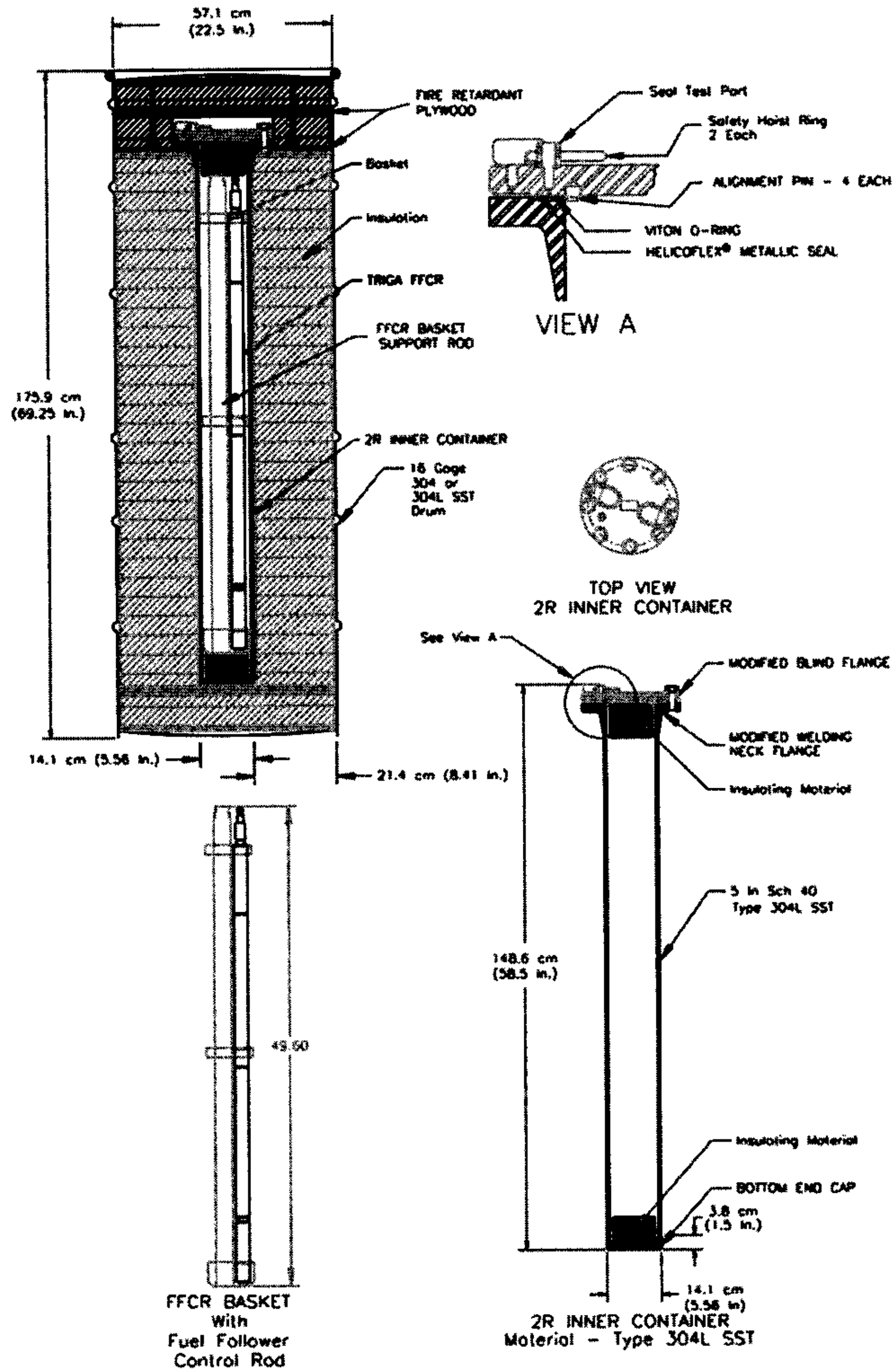


Figure D2-6. DOT-6M Container.



DOT 6M LEAK TESTABLE 420 LITER (110 GALLON) CONFIGURATION

Figure D2-7. NRF TRIGA Cask -
Fuel Configuration After Third Shipment.

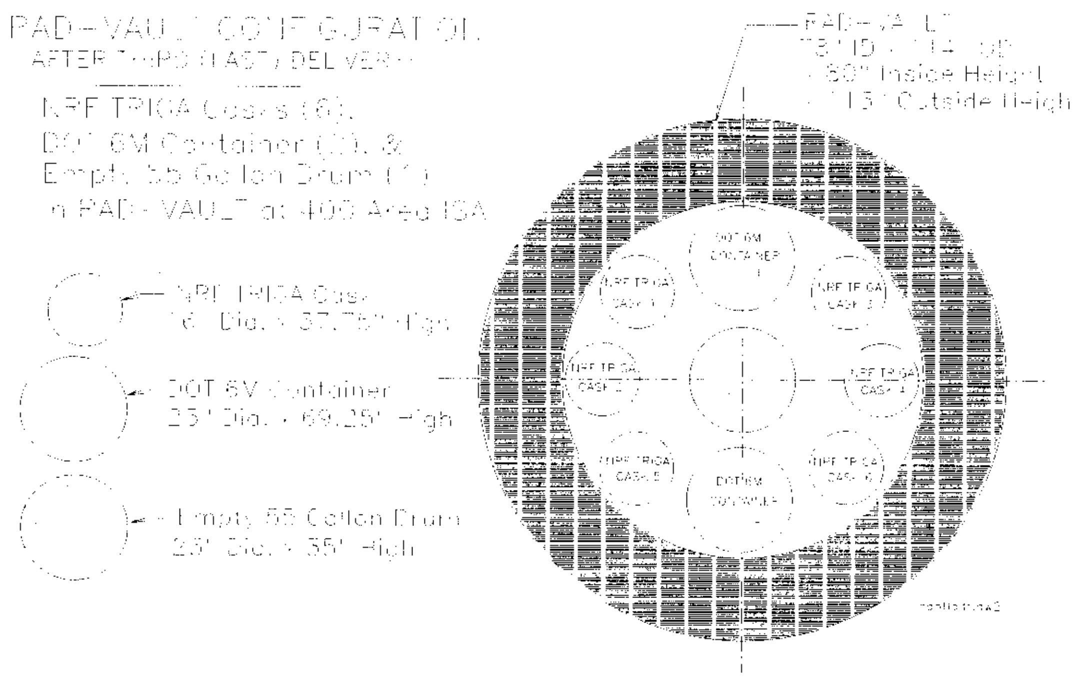


Figure D2-8. Interim Storage Cask.

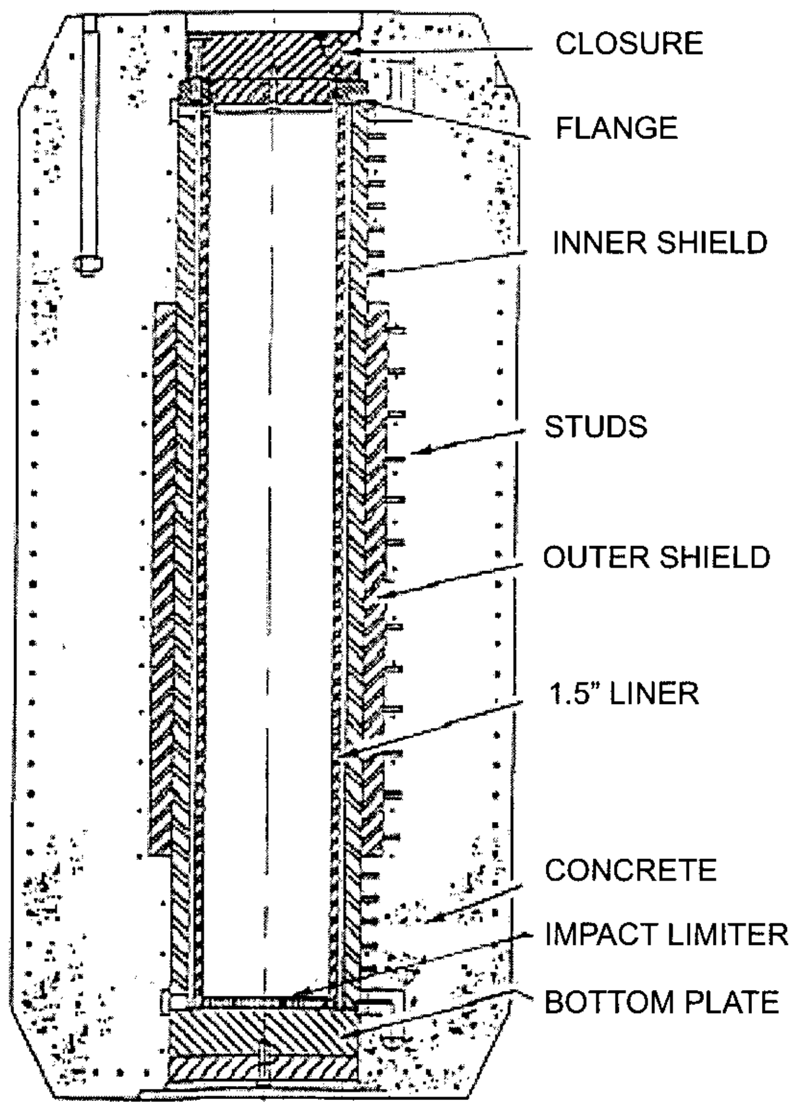


Figure D2-9. Core Component Container.

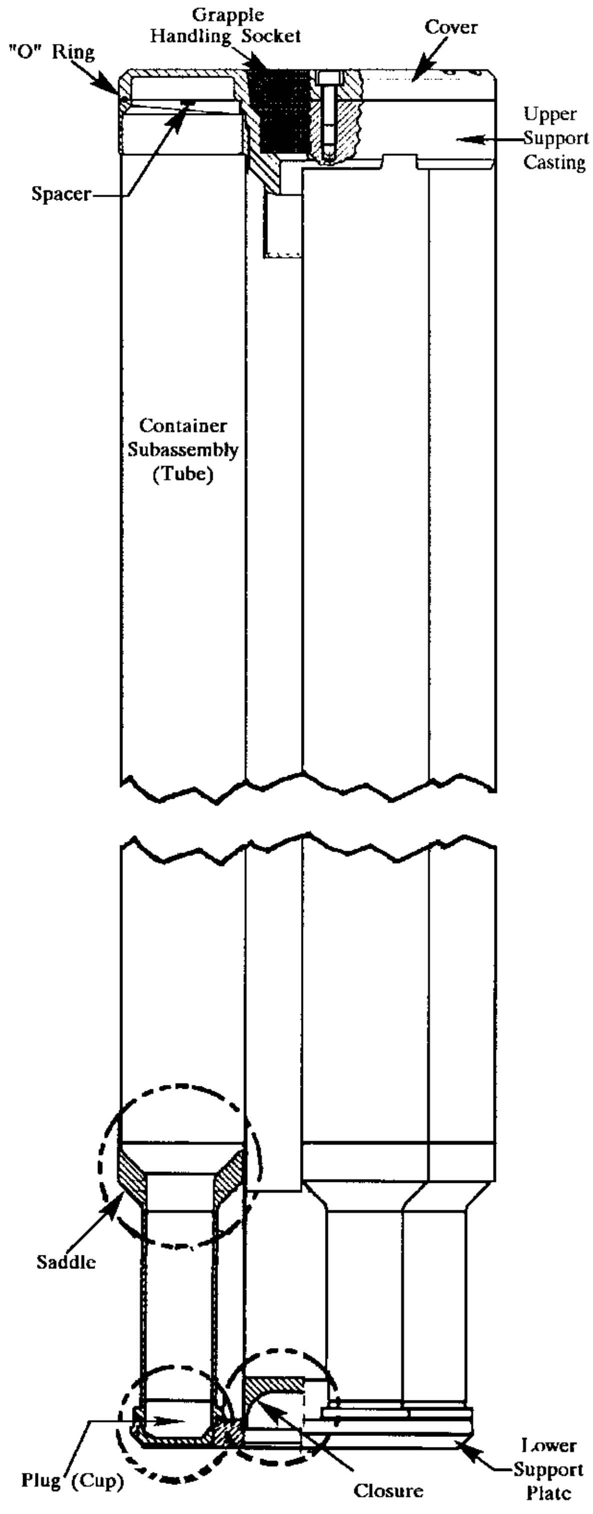


Figure D2-10. NAC-1 Cask.

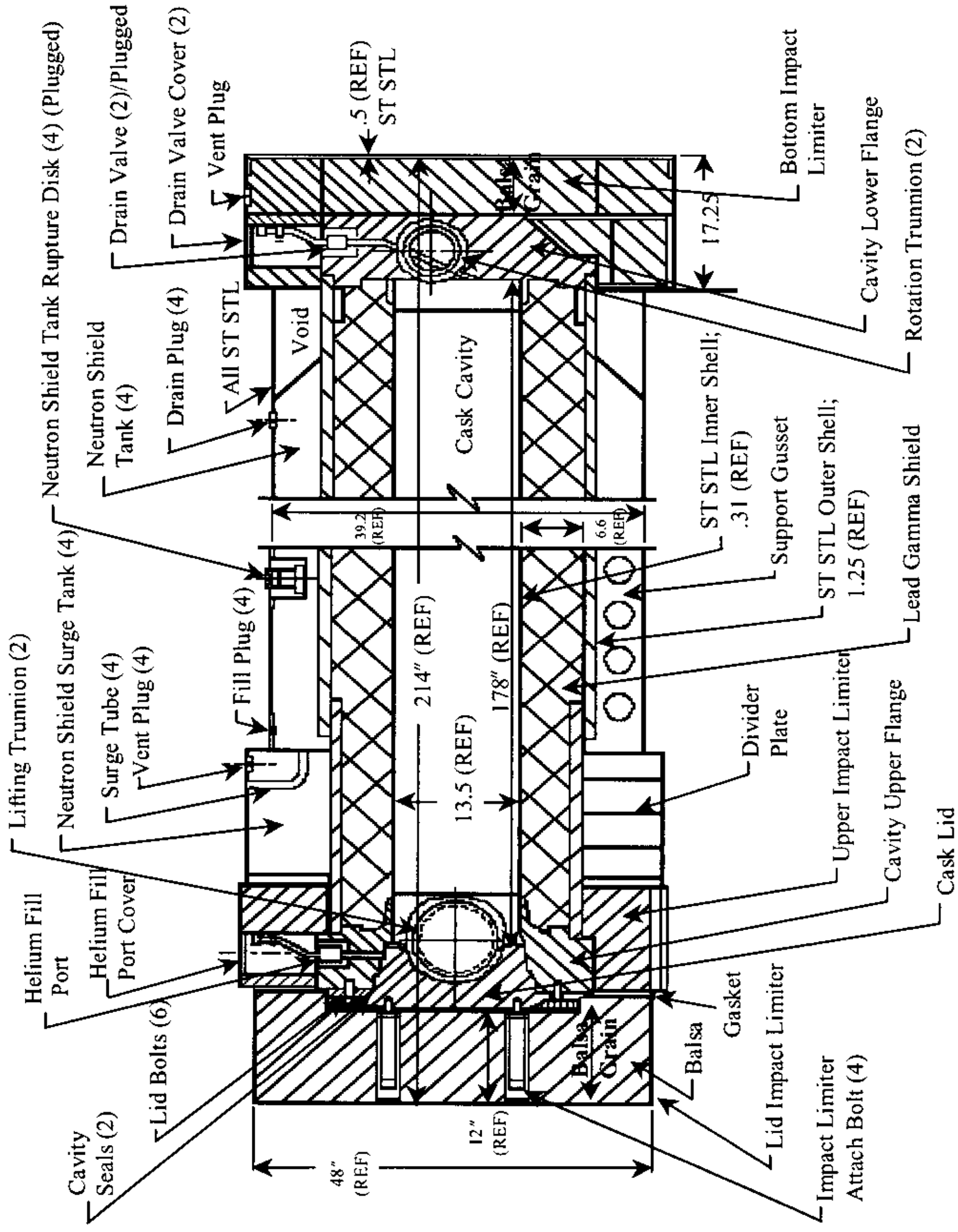


Figure D2-11. NAC-1 Cask/Light Water Reactor Canister System.

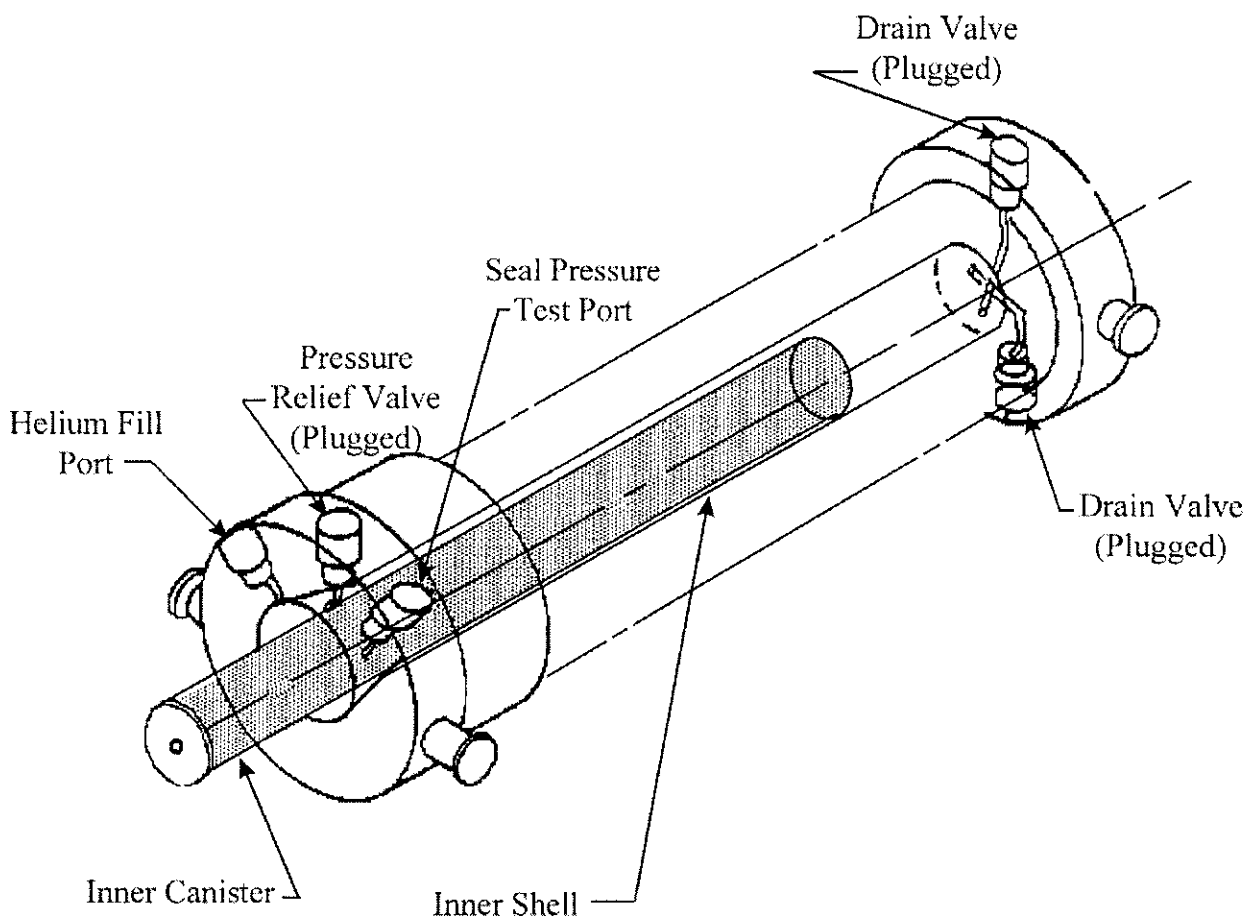


Figure D2-12. International Standards Organization Shipping Container - External Structure.

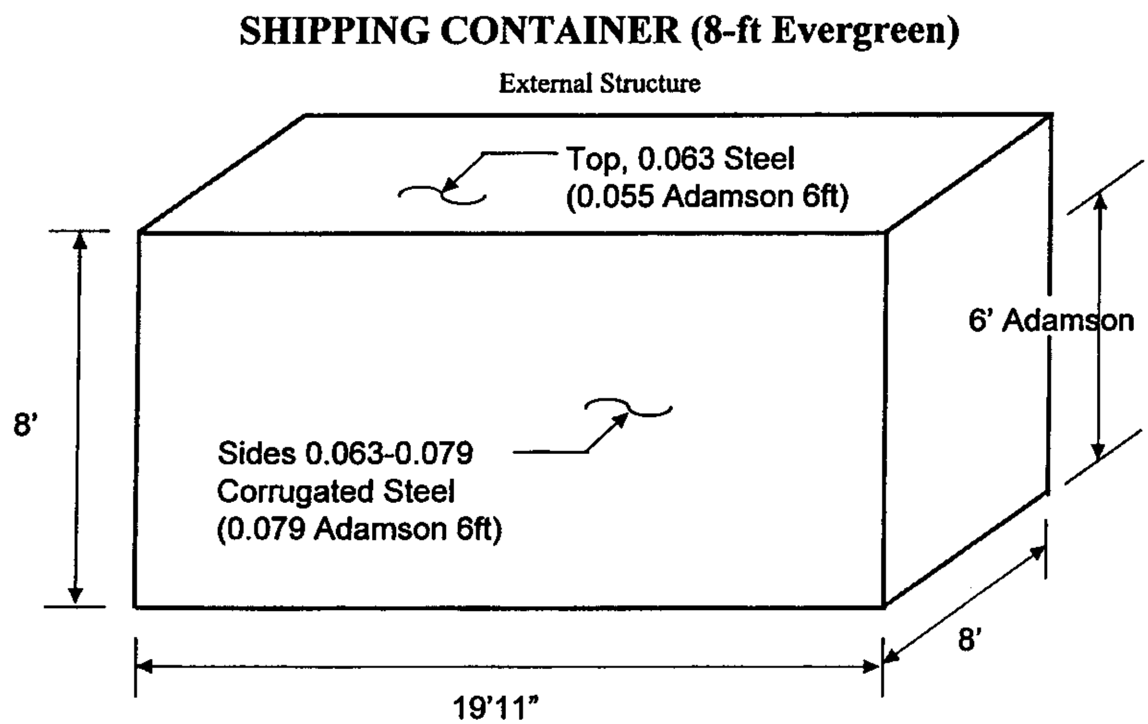


Figure D2-13. International Standards Organization Shipping Container - Internal Structure.

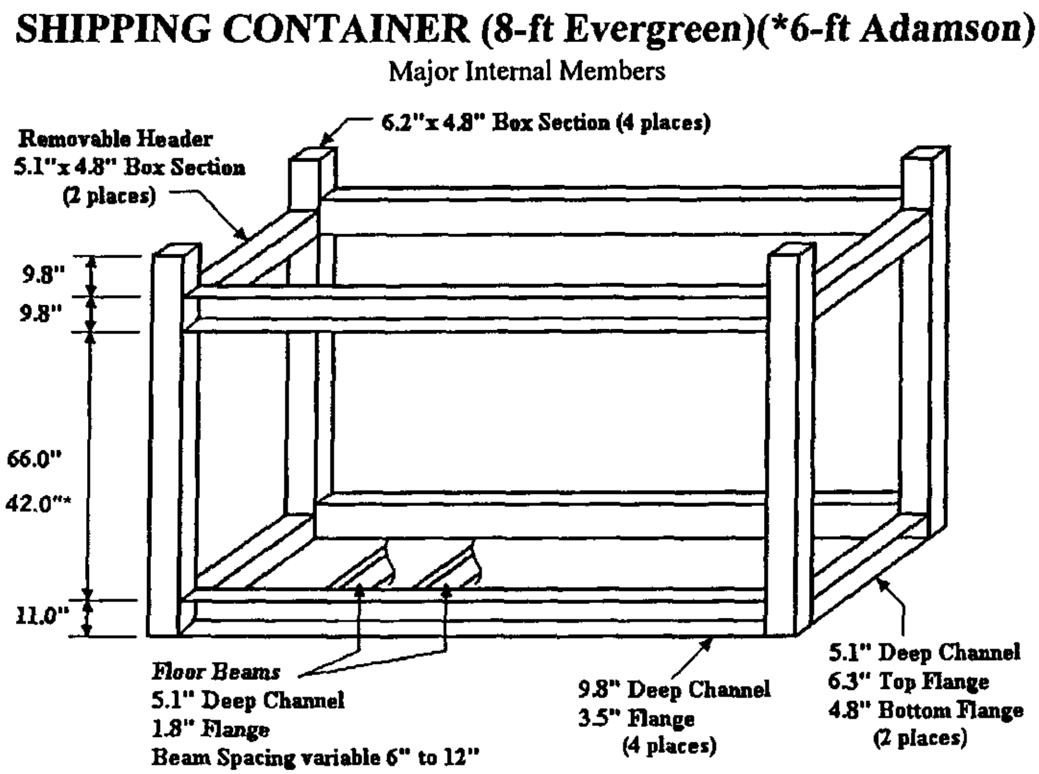


Figure D2-14. Light Water Reactor Fuel Inner Canister.

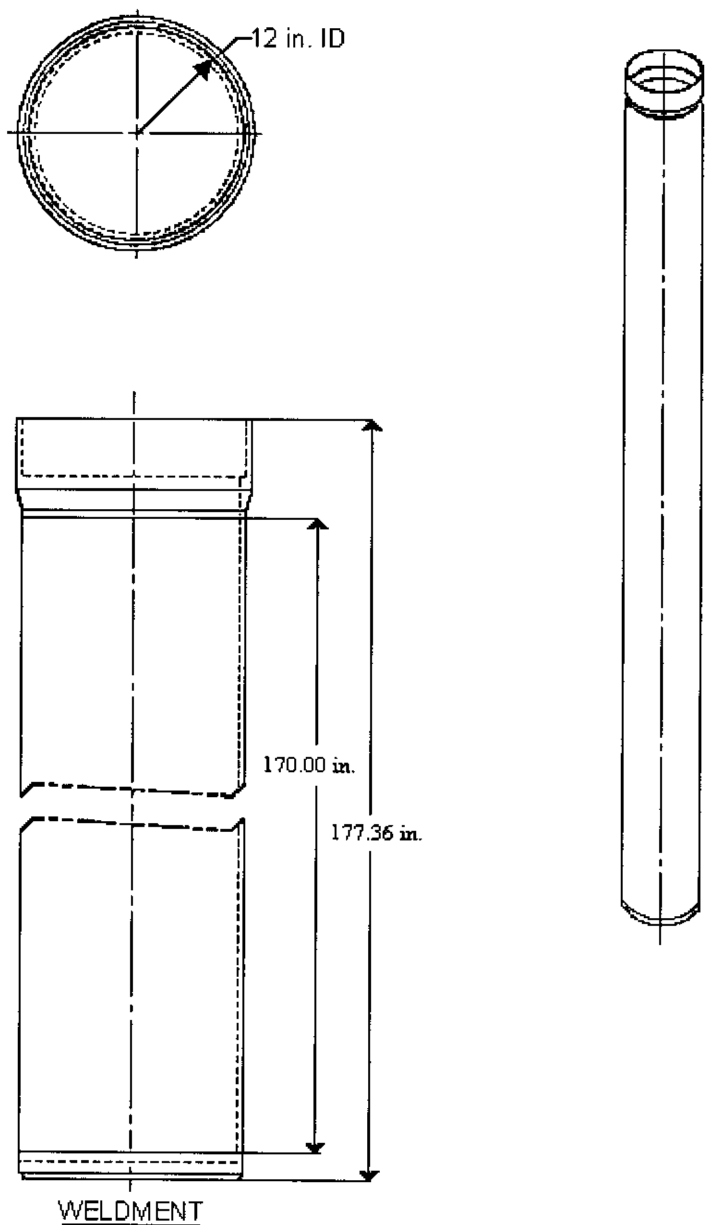
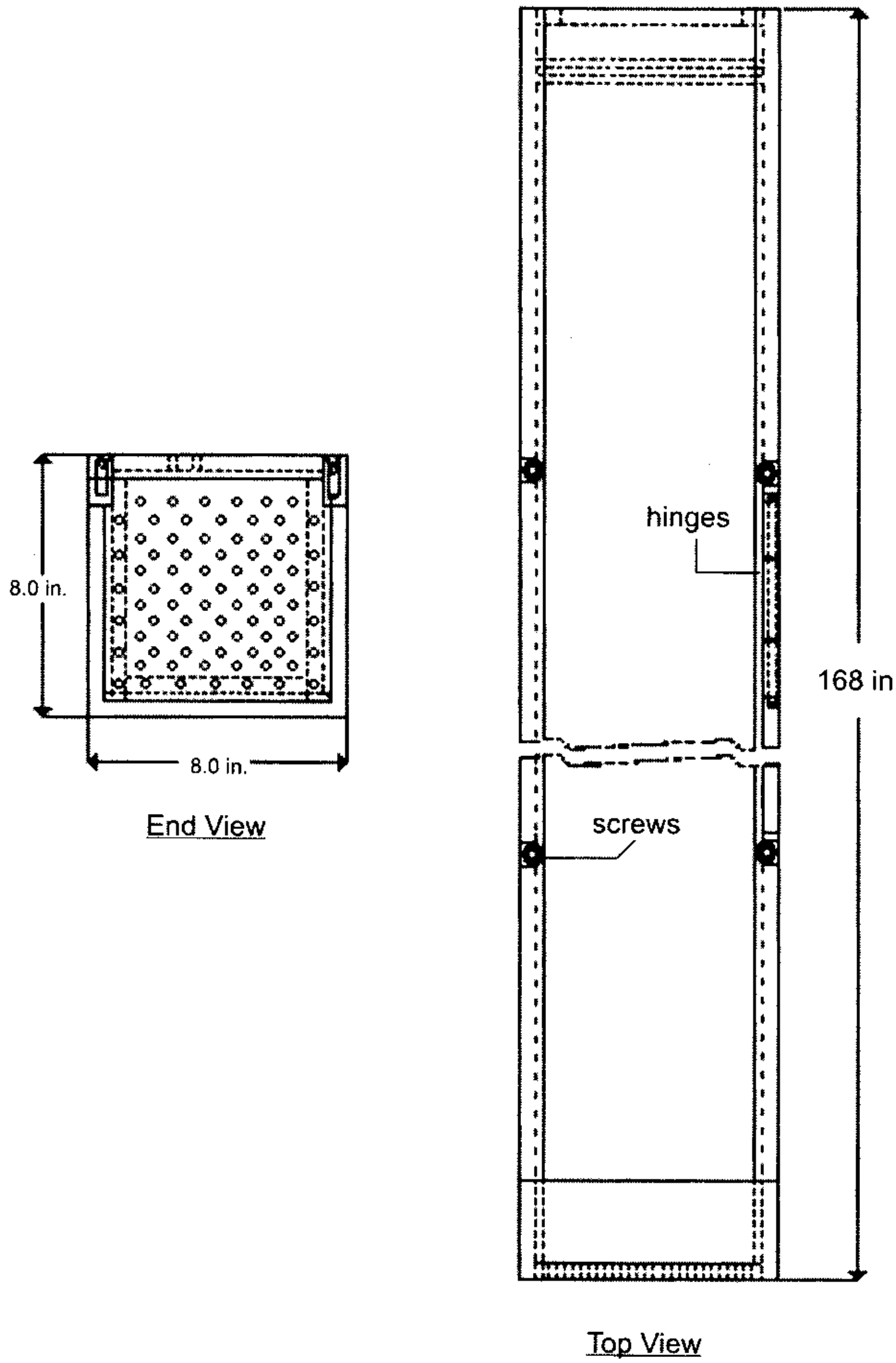


Figure D2-15. Loose Pin Consolidation Container.



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CHAPTER D3.0

HAZARD AND ACCIDENT ANALYSES

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CONTENTS

LIST OF TABLES

LIST OF TERMS

ALARA	as low as reasonably achievable
ASME	American Society of Mechanical Engineers
BDBA	beyond design basis accident
BWR	boiling water reactor
CCC	core component container
CEDE	committed effective dose equivalent
DBA	design basis accident
DBE	design basis earthquake
DBT	design basis tornado
DOE	U.S. Department of Energy
FFCR	fuel follower control rod
FFTF	Fast Flux Test Facility
FHA	fire hazard analysis
FSAR	final safety analysis report
ISA	interim storage area
ISC	interim storage cask
ISO	International Standards Organization
LWR	light water reactor
NRC	U.S. Nuclear Regulatory Commission
NRF	Neutron Radiography Facility
PWR	pressurized water reactor
SAR	safety analysis report
SNF	spent nuclear fuel
SSC	structure, system, and component
TSR	Technical Safety Requirement

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D3.0 HAZARD AND ACCIDENT ANALYSES

D3.1 INTRODUCTION

This chapter presents a summary of the key methodology, assumptions, and results of the final safety hazard analysis and design basis accident (DBA) analyses performed for the final design and operation of the 200 Area Interim Storage Area (ISA). These analyses form a safety basis for the final safety analysis report (FSAR) and present a comprehensive evaluation of the ISA handling- and storage-related activities and the natural phenomena and external hazards that can affect the public, workers, and environment. Single and multiple initiating events from equipment and human-error failures in the facility, and human and natural events outside of the facility, have been considered.

The contents of this chapter are as follows:

- The requirements for establishing the safety basis for the ISA are listed in Section D3.2. The requirements listed consist of U.S. Department of Energy (DOE) Orders and standards and applicable U.S. Nuclear Regulatory Commission (NRC) rules and guidance.

- The ISA hazard analysis is summarized in Section D3.3. The complete hazard analysis is contained in SNF-4820, *200 Area Interim Storage Area Final Hazard Analysis Report*. The analysis identifies hazard sources, hazardous conditions, potential accident scenarios and their initiators, and preliminary assessments of event frequencies and consequences. Hazards are identified by form and location and represent a complete spectrum of events that could occur throughout the facility. An initial set of safety features that would serve to prevent or mitigate the postulated accident scenarios is identified in the hazard analysis, with a final set of safety features identified in the accident analyses in Section D3.4.2 and in Chapters D4.0 and D5.0. Hazard analysis methodology and evaluation criteria are discussed in further detail in Section D3.3.

- The final facility hazard classification, determined in accordance with DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, is addressed in Section D3.3.2.2. The ISA has been assigned a final designation of a Hazard Category 2 facility based on material at risk.

- Section D3.3.2.3 contains discussions of defense in depth, worker safety, and environmental protection, including a detailed tabulation of engineered and administrative features that have been identified as providing for worker safety.

- High-risk events that pose a challenge to offsite release limits and onsite radiological dose evaluation guidelines have been selected from the hazard analysis for further detailed quantitative evaluation as DBAs. These DBAs are presented in Section D3.4.2.

Three receptor locations were used for the DBA analysis:

- Hanford Site boundary (17,390 m east of the ISA) — defined release limits; used for calculation of offsite doses and selection of safety-class features.
- Collocated worker (100 m from the ISA) — defined risk evaluation guidelines; used for calculation of onsite doses and selection of safety-significant features.
- Highway 240 (onsite, approximately 9,280 m west of the ISA) — no defined evaluation guideline; doses calculated for information purposes only (Scott 1995).

Safety-class and safety-significant features have been selected for each of the analyzed DBAs from the candidate structures, systems, and components (SSCs) identified in the hazard analysis. As required by the Spent Nuclear Fuel (SNF) Project's commitments to meet equivalent NRC requirements, candidate features have been identified as "important to safety" and are either designed, engineered, and procured consistent with safety-class or safety-significant classification requirements or in an NRC-equivalent manner appropriate for the planned contents. Safety-class and safety-significant features are presented first in Section D3.4.2, with the discussion of each accident, and described in more detail in Chapters D4.0 and D5.0.

Each of the DBAs that have been quantitatively analyzed represents a bounding case for a category or type of hazards and accidents. An in-depth review was performed of all significant accidents identified by the hazard analysis. The table and text that accompany each DBA in Section D3.4.2 include the preventive and mitigative features and the associated Technical Safety Requirements (TSRs) for the bounding case presented and for all other events binned within the accident category. Defense-in-depth features are also identified in these tables.

Chapter D3.0 interfaces with several other SNF Project safety documents. The planned scope and content of various other SNF Project safety reports, TSRs, and supporting safety documents are defined in Administrative Procedure MS-1-039, "Spent Nuclear Fuel Project ISMS Description." The hazards associated with transport and accident scenarios that are postulated during shipment are analyzed in the following safety analysis report for packaging documents:

- WHC-SD-TP-SARP-010, *Safety Analysis Report for Packaging (Onsite) Interim Storage Cask*
- NAC-E-804, *Safety Analysis Report for the NFS-4/NAC-1, Spent-Fuel Shipping Cask*
- WHC-SD-TP-SARP-008, *Safety Analysis Report for Packaging (Onsite) NRF TRIGA Packaging*
- HNF-8705, *Onsite Safety Analysis Report for Packaging – NFS-4/NAC-1 Spent Fuel Shipping Cask.*

D3.2 REQUIREMENTS

Chapter 3.0 of the HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1, lists the design codes, standards, regulations, and DOE Orders that contain requirements and guidance for establishing the safety basis for the SNF Project. Specific codes, standards, and requirements applicable to the 200 Area ISA are defined in HNF-SD-SNF-RD-001, *Spent Nuclear Fuel Project Standards/Requirements Identification Document*. Only the requirements that are specific for the ISA and that pertain to the safety analysis are provided here.

- Title 10, *Code of Federal Regulations*, Part 830, "Nuclear Safety Management" (10 CFR 830). This document has been verified to comply with the requirements of 10 CFR 830.

- DOE Order 5480.23, *Nuclear Safety Analysis Reports*, in conjunction with its Attachment 1, "Interim Guidance for DOE Order 5480.23." This document was written to comply with the requirements for analysis found in DOE Order 5480.23 by documenting the performance of hazard and accident analyses. The analyses were performed to the guidance of HNF-PRO-704, *Hazard and Accident Analysis Process*, which ensures that the hazards and accident analyses comply with DOE Order 5480.23. The methodology, assumptions, and criteria used to identify facility hazards, hazard rankings, candidate accidents, DBAs, preventive and mitigative features and controls, and the classification of these features (along with the definition of safety functions, performance criteria, and applicability) are described in Chapter 3.0 of HNF-3553, Volume 1, and documented in this chapter.

- DOE Order 5480.22, *Technical Safety Requirements*. This Order sets the requirements for developing and preparing a TSR document. This chapter complies with the requirements by documenting the performance of hazard and accident analyses in accordance with HNF-PRO-704 and DOE Order 5480.23. The results of the analyses were used to identify specific SSC safety functions, performance requirements for the SSCs, and the times for application of the safety functions.

- DOE Order 6430.1A, *General Design Criteria*. This Order provides requirements for the identification of safety-class items. The analyses documented in this chapter used the SSC classification requirements of DOE Order 6430.1A in the identification of safety-class SSCs. Compliance with DOE Order 6430.1A is demonstrated in SNF-5139, *200 Area Interim Storage Area DOE 6430.1A Compliance Evaluation*.

The following standards are used for content and guidance:

- DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*. This standard provides guidance for the preparation and review of hazard categorization and accident analysis techniques, as required by DOE Order 5480.23. Of particular importance to this chapter is the guidance pertaining to the hazard categorization methodology and the accident analysis techniques that are appropriate for the graded approach required by DOE Order 5480.23. Section D3.3.2.2 documents the final facility categorization of the ISA, determined in accordance with DOE-STD-1027-92.

- DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*. This standard supplements DOE Order 5480.23 by providing guidance specific to nonreactor nuclear facilities. In this regard, the standard provides detailed information on the performance of accident analyses for Hazard Category 2 and 3 facilities. The standard also provides guidance for establishing defense-in-depth measures and addresses safety-significant SSCs. This chapter has been organized and prepared consistent with the specifications of DOE-STD-3009-94.

In Letter 95-SFD-167, *Implementation of K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy* (Sellers 1995), DOE established the requirement for new SNF Project facilities to achieve "nuclear safety equivalency" to comparable NRC-licensed facilities. The SNF Project identified the NRC requirements that were to be met, in addition to existing and applicable DOE requirements, to establish nuclear safety equivalency. These NRC requirements, and the process used to identify them, are documented in HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements*, and in WHC-SD-SNF-DB-009, *Canister Storage Building Natural Phenomena Hazards*, Appendix C, "200 East Area Interim Storage Area Natural Phenomena Hazards." Applicable requirements have been imposed for the ISA.

The requirements for achieving NRC equivalency include the following:

- Title 10, *Code of Federal Regulations*, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR 72). This rule is used for licensing independent spent fuel storage installations. Section 72.122, "Overall Requirements," requires that the design bases for SSCs important to safety reflect appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. Section 72.24, "Content of Application: Technical Information," provides requirements in Paragraph 72.24(m) for the analysis of accidents and natural phenomena events that could result in a dose at the controlled area boundary.

- NRC Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)*. This guide establishes the format and content for safety analysis reports for license applications of fuel storage facilities.

Important-to-safety SSCs have been identified in accordance with 10 CFR 72.3. Once SSCs have been identified as having a function meeting the definition of important to safety, the requirements for SSCs important to safety specified in 10 CFR 72 are imposed. A graded approach is applied to an SSC important to safety by using the guidance provided in NUREG/CR-6407, *Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety*, as follows:

- Category A — Critical to Safe Operation

  SSCs in this category include those whose failure or malfunction could directly result in a condition adverse to public health and safety. Important-to-safety SSCs in this category are classified as safety class, as defined in DOE Order 6430.1A with the additional requirements therein.

- Category B — Major Impact on Safety

  SSCs in this category include those whose failure or malfunction could result in a condition adversely affecting collocated worker health and safety. Note that from the definition of Category C, Category B is understood to include events that could significantly damage the storage containers without severe impact to public health and safety. SSCs in this category are classified as safety significant, if they were not originally designed and procured in an NRC-equivalent manner appropriate for their planned contents.

- Category C — Minor Impact on Safety

  SSCs in this category include those whose failure or malfunction would not significantly reduce containment and would not be likely to create a situation adversely affecting public or collocated workers' health and safety. SSCs in this category are classified as general service.

While these documents have particular significance to this chapter, they do not, by themselves, establish requirements for the ISA or the Chapter D3.0 accident analyses. Requirements identified in HNF-SD-SNF-DB-003 and WHC-SD-SNF-DB-009 address NRC equivalency. It is noted that the SNF storage casks are "important to safety" based on the 10 CFR 72.3 definitions. The SNF Project has committed to comply with select NRC requirements. Additional requirements include safety documentation for (1) determining important-to-safety classifications for preventive and mitigative features; (2) identifying and resolving worker safety issues in DOE Order 6430.1A, DOE Order 5480.23, and DOE-STD-3009-94; and (3) evaluating nearby activities. The determination of important to safety is directly incorporated into the identified preventive or mitigative safety function features at the end of each DBA scenario in Section D3.4.2, and external human-generated threats are discussed in Section D3.3.2.3. Requirements of WHC-SD-SNF-DB-009 are incorporated into the natural phenomena hazard design criteria and are considered in the hazard evaluation.

Letter 9757710/97-SFD-172, *Contract No. DE-AC06-96RL13200 – Risk Evaluation Guidelines (REGs) to Ensure Inherently Safer Designs* (Sellers 1997), provides another SNF Project requirement significant in the development of this chapter (e.g., safety-significant SSCs).

D3.3 HAZARD ANALYSIS

The hazard identification and evaluation process provides a thorough, predominantly qualitative evaluation of the spectrum of risks to the public, workers, and the environment caused by accidents that involve the hazards identified in the hazard analysis. This process is further described in Section 3.3 of HNF-3553, Volume 1. HNF-2015, *200 Area Interim Storage Area Preliminary Hazard Analysis Report*, was prepared to support HNF-2524, *200 East Area Interim Storage Area Preliminary Safety Evaluation Report*, and to integrate all previous ISA hazard analyses. A final hazard analysis has been performed to support the accident analysis and is documented in SNF-4820. The final hazard analysis systematically reviewed the final ISA design, as described in Chapter D2.0, associated supporting design documentation, and design references to identify any additional hazardous materials or energy sources that have the potential to initiate an accident that could require further review or analysis. This process resulted in the selection of seven DBAs for more comprehensive analysis in Section D3.4.2.

D3.3.1 Methodology

The methodology used to identify and evaluate the SNF Project facility hazards is described in detail in Section 3.3 of HNF-3553, Volume 1. This section discusses the areas of this methodology that are specific to the ISA. The hazard evaluation process identified hazardous conditions, determined causes and preventive and mitigative features, and qualitatively estimated the consequences and frequencies of occurrence. The results of the application of this methodology to the ISA are presented in Section D3.3.2. The hazard analysis was performed in accordance with DOE-STD-3009-94 and implements the requirements of DOE Order 5480.23.

D3.3.1.1 Hazard Identification.

No adverse consequences for the public or workers, or that cause contamination of the environment, are expected under normal operations or storage. The hazard analysis focused on abnormal and accident conditions, as described in SNF-4820. The hazards associated with transport and the accident scenarios that are postulated during shipment are analyzed in the safety analysis reports for packaging.

The ISA cask handling and storage activities that can take place within the boundary of the storage area were identified, and the hazards were identified by form (e.g., electrical, thermal, friction), type (e.g., motors, power tools, wiring), and location. A standardized hazardous material/energy source checklist (see Section 3.3.1.1, Table 3-2, of HNF-3553, Volume 1) was used to group the potentially hazardous materials and energy sources. The methodology of the hazards identification process is described further in the Section 3.3.1.1 of HNF-3553, Volume 1.

D3.3.1.2 Hazard Evaluation.

As described in Section D3.3.1.1, hazards associated with abnormal and accident conditions at the ISA were identified. The hazards identified were evaluated to determine the causes of the hazard, potential accidents that could result from the presence of each hazard, and consequences to the public offsite, the collocated and facility workers onsite, the environment, or the ISA. Safety features, segregated into preventive and mitigative features, were identified for each hazard based on the ability of the feature to prevent or mitigate the consequences. Qualitative estimates of the frequency and consequences of the hazardous condition also were assigned (see Section 3.3.1.2 of HNF-3553, Volume 1, for the criteria used in assigning the consequence and frequency categories).

D3.3.2 Hazard Analysis Results

The hazard analysis process is described in SNF-4820 and summarized in Section D3.3.1 and in Section 3.3.1 of HNF-3553, Volume 1. The products of that process, in the order of progression, are as follows:

1. Develop a series of checklist-style tables describing hazardous materials and energy sources, organized by fuel type. These tables were used to develop hazard analysis accident scenarios.
2. Develop a series of tables describing the standard industrial hazards considered, organized by fuel type. These events were judged to have no contribution to uncontrolled radiological and/or hazardous materials releases and were not considered in the selection of DBAs, safety-class or safety-significant features, or TSRs. They were among the hazards considered and, therefore, are included in this analysis for completeness.
3. Develop a series of tables describing potential hazard scenarios. These tables include hazardous energy sources and materials, hazardous conditions, causes and initiators, potential accidents, qualitative determinations of event frequencies and consequences, safety features for prevention and/or mitigation of the consequences, and defense-in-depth or worker safety features.
4. Compile a table assigning risk bins to causes associated with significant consequences to offsite and onsite receptors. Consistent with DOE-STD-3009-94, the events located in risk bins representing "situations of concern" or "situations of major concern" were evaluated as candidate DBAs.
5. Prepare a final list of candidate DBAs sorted by risk ranking and energy change or release. This list formed the basis for selection of the DBAs presented in Section D3.4.2.

DBA selection is addressed in Section D3.3.2.3.5 and its accompanying Table D3-5. In terms of the risk binning process, the accidents chosen from the hazard analysis for further analysis as DBAs were all events identified in consequence categories S3 and S2. These categories indicate significant effects to offsite and onsite receptors.

The final hazard analysis has been reconciled with the up-to-date ISA facility design described in Chapters D2.0, D4.0, and D5.0.

D3.3.2.1 Hazard Identification.

The final ISA hazard analysis tables are shown in SNF-4820. The main inventory of hazardous material in the ISA is the radionuclide content of the stored fuel. The toxicological hazards of the radionuclide inventory were reviewed. As described in Section D3.4.1.1, the radiological guidelines are more limiting than the toxicological guidelines for the release of SNF particulate. Other hazardous material identified by the hazard identification process includes pyrophoric metals and hydrides, sodium, oxidizers, hydrogen, diesel fuel, and other flammable or combustible materials. A specific and comprehensive analysis of all fire hazards associated with the ISA was completed to augment the standard hazard analysis and is documented in SNF-4932, *Fire Hazards Analysis for the 200 Area Interim Storage Area.*

The ISA does not have an operating history; so major hazards resulting from facility operation cannot be identified or summarized as suggested by DOE-STD-3009-94. However, as described in this section, the ISA spent fuel handling and storage activities are similar to those used by the independent spent fuel storage installations that were issued licenses under 10 CFR 72. These hazards include generation of combustible gases, failure of confinement barriers, defects in cask integrity, and the spread of external contamination.

D3.3.2.2 Hazard Classification.

A final hazards categorization of the ISA facility was performed based on the final hazard analysis (SNF-4820) and accident analyses documentation for the facility. Consistent with DOE-STD-1027-92, the final categorization was based on the material-at-risk quantities identified in an individual cask inventory, as described in Section D3.4.1.1. The ISA material-at-risk quantities were compared against the DOE-STD-1027-92 threshold quantities. The ISA facility final hazard categorization found the ISA facility to be a Hazard Category 2 facility. This categorization level is consistent with the bases and guidance described in DOE-STD-1027-92.

The radioactive inventory for the 200 Area ISA is established in HNF-1755, *Initial Hazard Classification for the 200 Area Interim Storage Area, Project W-518*. HNF-1755 defines an inventory for safety analysis based on selecting the high-burnup fuel and fuel type that would result in the highest estimated dose to people exposed to the material, and then treating all the ISA fuel types as high-burnup fuel. The radioactive inventory for each of the ISA fuel types, as identified in HNF-1755, is presented in Table D3-1, Table D3-2, and Table D3-3 for Fast Flux Test Facility (FFTF) SNF; Neutron Radiography Facility (NRF) TRIGA[1] SNF; and light water reactor (LWR) SNF, respectively.

The established safety basis radiological nuclide inventory of each fuel type was used to estimate the material quantities available for release by multiplying the material at risk and the specific nuclide inventories for each fuel type. The material at risk was taken as the amount of fuel stored in an individual cask for each fuel type. These quantities were compared against the Category 2 threshold values from DOE-STD-1027-92, Table A.1 (see Table D3-1, Table D3-2, and Table D3-3). The sum of the material inventory/Category 2 quantity thresholds is 234 for the FFTF fuel, 0.03 for the NRF TRIGA fuel, and 82.2 for the commercial LWR fuel. Since the sum of ratios is greater than one when compared to Category 2 threshold criteria for both the FFTF and LWR fuel, the ISA remains a Category 2 facility as identified during the preliminary hazard classification.

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[1] TRIGA is a trademark of General Dynamics Corporation.

Table D3-1. Final Hazard Category Comparison for Fast Flux Test Facility Fuel Inventory per Cask at the 200 Area Interim Storage Area.

Radionuclide	FFTF SNF inventory (single assembly)	FFTF SNF inventory (7 assemblies)	Category 2 threshold curies	Ratio of facility inventory to Category 2 curies
H-3	78.2	547	300,000	0.00
Kr-85	674	4,718	28,000,000	0.00
Sr-90	5,040	35,280	22,000	1.60
Y-90	5,050	35,350	430,000	0.08
Tc-99	1.99	14	3,800,000	0.00
Ru-106	4,890	34,230	6,500	5.27
Ag-110m	8.89	62	530,000	0.00
Cd-113m	12.9	90	430,000	0.00
Sb-125	1,210	8,470	430,000	0.02
I-129	0.00663	0	430,000	0.00
Cs-134	2,510	17,570	60,000	0.29
Cs-137	13,700	95,900	89,000	1.08
Ce-144	1,660	11,620	82,000	0.14
Pr-144	1,660	11,620	430,000	0.03
Pm-147	8,730	61,110	840,000	0.07
Sm-151	521	3,647	990,000	0.00
Pu-238	281	1,967	62	31.73
Pu-239	349	2,443	28	87.25
Pu-240	268	1,876	55	34.11
Pu-241	14,600	102,200	2,900	35.24
Am-241	253	1,771	55	32.20
Am-242m	19.3	135	56	2.41
Cm-244	20.4	143	55	2.60
Sum of the ratios				234.12

FFTF = Fast Flux Test Facility.
SNF = spent nuclear fuel.

Table D3-2. Final Hazard Category Comparison for Neutron Radiography Facility TRIGA Fuel Inventory per Cask at the 200 Area Interim Storage Area.

Radionuclide	TRIGA[a] SNF inventory (single assembly)	TRIGA[a] SNF inventory (101 assemblies)	Category 3 threshold curies	Category 2 threshold curies	Ratio of facility inventory to Category 3 curies	Ratio of facility inventory to Category 2 curies
H-3	0.005156	0.52	1,000	300,000	0.00	0.000
C-14	0.00001209	0.00	420	1,400,000	0.00	0.000
Fe-55	0.1904	19.23	5,400	11,000,000	0.00	0.000
Ni-59	0.0005488	0.06	11,800[b]	430,000	0.00	0.000
Ni-63	0.05943	6.00	5,400	4,500,000	0.00	0.000
Co-60	0.0006458	0.07	280	190,000	0.00	0.000
Zr-93	0.00002648	0.00	62	89,000	0.00	0.000
Kr-85	0.1273	12.86	20,000	28,000,000	0.00	0.000
Sr-90	1.864	188.26	16	22,000	11.77	0.009
Y-90	1.865	188.37	1,420[b]	430,000	0.13	0.000
Tc-99	0.0004061	0.04	1,700	3,800,000	0.00	0.000
Ru-106	0.03391	3.42	100	6,500	0.03	0.001
Rh-106	0.03391	3.42	1,960[b]	430,000	0.00	0.000
Cd-113m	0.0001943	0.02	11.8[b]	430,000	0.00	0.000
Sb-125	0.01368	1.38	60[b]	430,000	0.02	0.000
Te-125m	0.003337	0.34	36[b]	430,000	0.01	0.000
Cs-134	0.009104	0.92	42	60,000	0.02	0.000
Cs-137	1.965	198.47	60	89,000	3.31	0.002
Ce-144	0.1863	18.82	100	82,000	0.19	0.000
Pm-147	0.6730	67.97	1,000	840,000	0.07	0.000
Sm-151	0.04990	5.04	1,000	990,000	0.01	0.000
Eu-152	0.002756	0.28	200	130,000	0.00	0.000
Eu-154	0.002114	0.21	200	110,000	0.00	0.000
Eu-155	0.01505	1.52	940	730,000	0.00	0.000
U-234	0.002350	0.24	4.2	220	0.06	0.001
U-235	0.00007969	0.01	4.2	110,000,000	0.00	0.000
U-238	0.00005173	0.01	4.2	240	0.00	0.000
Np-237	0.000001769	0.00	0.42	58	0.00	0.000
Pu-238	0.0003155	0.03	0.62	62	0.05	0.000
Pu-239	0.004455	0.45	0.52	28	0.87	0.016
Pu-240	0.0002871	0.03	0.026[b]	55	1.15	0.001
Pu-241	0.002637	0.27	32	2,900	0.01	0.000
Am-241	0.00005746	0.01	0.52	55	0.02	0.000
Sum totals					20.72	0.030

[a] TRIGA is a trademark of General Dynamics Corporation.

[b] Category 3 threshold quantities for these radionuclides are not presented in DOE-STD-1027-92, (Change Notice No. 1-1997), *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*. These values are taken from WHC-SD-GN-HC-20002, 1996, *Category 3 Threshold Quantities for Hazard Categorization of Nonreactor Facilities*, Rev. 0.

SNF = spent nuclear fuel.

Table D3-3. Final Initial Hazard Category Comparison for Light Water Reactor Fuel Inventory per Cask at the 200 Area Interim Storage Area.

Radionuclide	LWR fuel inventory (single assembly)	Category 2 threshold curies	Ratio of facility inventory to Category 2 curies
H-3	92.8	300,000	0.00
Co-60	788	190,000	0.00
Kr-85	1,310	28,000,000	0.00
Sr-90	19,800	22,000	0.90
Y-90	19,800	430,000	0.05
Tc-99	5.21	3,800,000	0.00
Rh-106	8.43	430,000	0.00
Ru-106	8.42	6,500	0.00
Cd-113m	13.1	430,000	0.00
Sb-125	130	430,000	0.00
Te-125m	31.7	430,000	0.00
I-129	0.0126	430,000	0.00
Cs-134	491	60,000	0.01
Cs-137	28,500	89,000	0.32
Pm-147	817	840,000	0.00
Sm-151	123	990,000	0.00
Eu-154	1,550	110,000	0.01
Eu-155	402	730,000	0.00
U-234	0.472	220	0.00
U-235	0.00691	240	0.00
U-236	0.107	55	0.00
U-238	0.130	240	0.00
Np-237	0.134	58	0.00
Np-239	12.8	55	0.23
Pu-238	1,360	62	21.94
Pu-239	125	28	4.46
Pu-240	212	55	3.85
Pu-241	24,400	2,900	8.41
Pu-242	0.939	55	0.02
Am-241	944	55	17.16
Am-242	10.7	55	0.19
Am-242m	10.7	56	0.19
Am-243	12.8	55	0.23
Cm-242	8.82	1,700	0.01
Cm-243	10.5	55	0.19
Cm-244	1,210	55	22.00
Sum of the ratios			82.17

LWR = light water reactor.

D3.3.2.3 Hazard Evaluation.

The final ISA facility hazard analysis identified hazards associated with handling and storage operations to be used in the ISA. Standard industrial hazards were identified and removed from the list of facility hazards used to identify the DBAs. The results of the hazard analysis identified hazard scenarios in the ISA. These scenarios were used to define the ISA DBAs selected for further analysis in Section D3.4.2.

The external hazards from human-generated threats to ISA operation identified in Section D1.6 involve only those from aircraft activity. As reported in HNF-1786, *Assessment of Aircraft Impact Frequency for the 200 Area Interim Storage Area*, nine active airports within a 24-mile radius of the ISA define a maximum credible evaluation basis aircraft impact for the ISA. Therefore, the ISA facility is adequately protected from this external hazard.

Evaluation of the hazards of nearby external human-generated activities that could represent a threat to the facility is an NRC-equivalency requirement that is specified in HNF-SD-SNF-DB-003. Section D1.7 identifies and addresses the threats to the ISA from nearby external facilities. As stated in Section D1.6, other threats to ISA operation from external human-generated activities that are not currently well known at this time will be evaluated by the unreviewed safety question process.

The ISA fire hazard analysis (SNF-4932) identified the fire hazards, fire loading criteria, and appropriate requirements for addressing fire hazards during ISA operation based on applicable DOE Orders and regulations. The findings of the ISA fire hazard analysis were incorporated into the design and engineering of ISA SSCs and controls (e.g., fire loads, inspections, and watches). Therefore, credible fire hazards are addressed in the design of the facility. Threats from nearby facilities are identified in Section D1.7 based on applicable DOE Orders and regulations. Credible threats from nearby facilities are addressed in Section D1.7.

D3.3.2.3.1 Planned Design and Operational Safety Improvements.

This section discusses commitments for planned, but not yet implemented, major design, handling, or storage improvements for the facility. However, there are currently no outstanding major improvements for improved safety identified or planned as a result of the hazard evaluation that are not part of the current design and planned facility operations.

D3.3.2.3.2 Defense in Depth.

A summary of fundamental points relevant to the concept of defense in depth are described in Section 3.3.2.3.2 of HNF-3553, Volume 1.

Features Chosen to Provide Defense in Depth for the 200 Area Interim Storage Area

Defense-in-depth features for the ISA were selected based on a relative ranking of the hazards from the hazard identification process, followed by selection of the safety-class and safety-significant features and TSRs for the DBAs, which are described in Section D3.4.2. Preventive and mitigative features identified in the hazard analysis (SNF-4820) but not identified in the accident analysis as safety-class, safety-significant, or TSRs are identified as additional defense-in-depth features. The defense-in-depth features are presented in the tables that accompany each DBA in Section D3.4.2. The NRF TRIGA cask and DOT-6M[2] container were considered for upgrade to safety significant for their contribution to defense-in-depth for multiple accidents, but the containers were not selected for upgrade. These general-service containers were originally designed and procured in an NRC-equivalent manner and have been shown by analysis to provide sufficient confinement and protective functions for the low activity TRIGA fuel. These items are currently in use at the 400 Area ISA. Administrative features identified in these tables are in addition to those already identified in Chapters D4.0 and D5.0, and the programmatic chapters of this Annex (e.g., Chapters D7.0, D8.0, and D11.0).

All SSCs are designed in accordance with applicable codes and standards with a high degree of reliability and simplicity, and the design encompasses human factors considerations to ensure that operations can be conducted safely. Defense-in-depth features for preventing and mitigating hazards and accidents have also been identified. An abnormal or accident cask is defined as one received out of specification or one that is dropped, impacted, or damaged at the ISA. Such casks will be handled using recovery operations under operations-related procedures. Recovery will be based on analysis and the development of a recovery plan by an appropriately qualified recovery team.

Safety-Significant Structures, Systems, and Components

Safety-significant SSCs are predominantly required to prevent or mitigate consequences of postulated accident events to the collocated onsite worker. In addition, DOE-STD-3009-94 suggests that SSCs be designated as safety significant, if they play a key role in defense in depth or worker safety. The severity of the event being prevented or mitigated, and the number of barriers present, are provided in DOE-STD-3009-94 as guidance for the identification of defense-in-depth safety-significant SSCs.

Technical Safety Requirements

TSRs were identified for postulated accident events that could challenge accident consequence release limits and evaluation guidelines for the offsite public and collocated onsite worker. These TSRs are identified in the individual DBA sections and further explained in Chapter D5.0. In addition, criticality prevention features are controlled by the TSRs, as identified in Chapter 5.0 of HNF-3553, Volume 1, and in Chapters D5.0 and D6.0. No defense-in-depth features are identified as requiring TSR coverage.

D3.3.2.3.3 Worker Safety.

Worker safety for the ISA is ensured by a combination of design features that reduce exposure to radioactive, toxic, and industrial hazards, and by institutional practices that, in total, provide protection of workers from these hazards. Protection of the facility worker from the standard industrial hazards identified for the ISA is achieved through adherence to the institutional safety programs described in Chapters 7.0, 8.0, 9.0, 11.0, 15.0, and 17.0 of HNF-3553, Volume 1, and documented in lower-tier documents (e.g., health and safety plans

[2] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.

and job hazards analyses). These industrial hazards do not require specific safety-significant SSCs or TSR-level administrative features. Therefore, in accordance with the guidance of DOE-STD-3009-94, the remainder of this section deals with protecting workers from the hazards of facility operation that are exclusive of standard industrial hazards.

The final ISA hazard analysis provides an overview of the major features that protect facility workers at the ISA (SNF-4820). Worker safety features are an integral part of facility design and operation. The major features of worker protection are identified in Table D3-4 and are categorized by hazard. The features presented in Table D3-4 are in addition to those identified as safety-class or safety-significant features in the DBA sections. The hazard energy source or material and hazardous condition are identified along with protective worker safety features, which include passive, active, and administrative features. SSCs are identified as safety significant for the ISA based on their ability to prevent or mitigate serious impacts to worker safety.

D3.3.2.3.4 Environmental Protection.

The hazard to the environment from ISA operations involves the potential release of contaminants. The release pathway for these contaminants is only via the air to the boundaries and receptors discussed in Section D1.3.1.3. No liquid release hazards or accidents have been identified, and no contaminant releases to the ground or groundwater are involved for the ISA.

Based on the ISA design and operating information, no use of toxic chemicals has been identified. The toxicological hazards of the radionuclide inventory have been reviewed. As described in Section D3.4.1.1, the radiological guidelines are more limiting than the toxicological guidelines for any release from the material stored in the ISA. Implementation of the prevention and mitigation features will prevent large releases that could have significant environmental impact.

The project features that protect the onsite collocated worker and the offsite public against radiological exposure also serve to prevent and mitigate radiological release to the environment. In addition, sitewide programs for environmental monitoring provide for assessment of the impact of facility releases. Normal ISA handling or storage activities are expected to have a minor impact on the local and regional environment.

D3.3.2.3.5 Accident Selection.

The methodology for the selection of DBAs is specified in DOE-STD-3009-94. DBAs are to be selected so that the range of accident scenarios analyzed in the accident analysis represents a complete set of representative and bounding conditions. This is a common requirement among the SNF Project facilities and is described in Section 3.3.2.3.5 of HNF-3553, Volume 1. The list of candidate accidents resulting from the hazards binning process for the ISA facility is presented in Table D3-5. The table contains all identified Category S3 and S2 events. All of these events, and the controls selected for their prevention and mitigation, are described in Sections D3.4.2.1 through D3.4.2.7.

Table D3-4. Safety Features for Facility Workers (S1 Consequence Items). (4 sheets)

Location/ checklist entry	Hazardous condition	Potential accident	Worker consequences	Features in addition to those identified in the design basis accident sections	
				Engineering features	Administrative features
Storage area/ B-07 B-10 B-11	Cask overheating Hot surfaces Hot gases	Exceeding the design temperature of the cask Loss of confinement of the cask Pressurization of the cask	Worker exposure to radioactive particulate Personnel injury	Passively cooled cask design; cask designed to withstand temperatures in excess of any of those anticipated. Designed for internal pressure to withstand worst possible scenario. Cooling channel in ISC.	Procedures define the types of personnel protective equipment. Low burnup TRIGA[a] decay heat loads. LWR fuel lower burnup than NAC-1[b] design. TSR: Cask spacing requirements on the location of the casks. White paint on the outside of the cask. Note: No temperature monitoring.
Storage area/ D-02 D-06	Loss of shipping container structural integrity	Drop of container upon lifting	Personnel injury	Paint coating on container. NAC-1[b] channels on bottom of container.	Routine inspection and maintenance.
Storage area/ F-01 F-02 F-04 F-05	Vehicle collision with cask Vehicle collision with other equipment	Vehicle collision with: • Casks • Equipment • Obstructions Loss of confinement	Worker exposure to radioactive particulate	Cask designed to withstand impact.	Fenced perimeter and locked gates. Restricted access inside fenced area.

Table D3-4. Safety Features for Facility Workers (S1 Consequence Items). (4 sheets)

Location/ checklist entry	Hazardous condition	Potential accident	Worker consequences	Features in addition to those identified in the design basis accident sections	
				Engineering features	**Administrative features**
Storage area/ G-03 G-06 G-13	A drop of the boom on the cask A drop of the cask while being moved by the crane	Cask is breached Cask falls from crane to pad or transporter Cask is overturned	Personnel injury Worker exposure to radioactive particulate	Hoist designed for single failure. ISC survives 4-ft drop to unyielding surface. DOT-6M[c] cask. TRIGA[a] cask designed to withstand a 109-in. drop to concrete. For NAC-1[b] cask, upper and lower impact limiters.	TRIGA[a] Rad-Vault[d] will not be picked up when full. Compliance with hoisting and rigging manual. TSR: Limit height to which the cask is picked up. Qualified crane operators. Detailed procedures.
Storage area/ H-06 H-07 H-11	Overpressuri-zation of cask and release of gases from cask	Loss of confinement and release of radioactive gases and particulate	Personnel injury Worker exposure to radioactive particulate Personnel contamination	Casks designed to withstand pressurization.	Casks are inspected.
Storage area/ J-06 L-11	Flammable hydrogen-air mixture and an ignition source causing combustion	Deflagration may cause excessive pressure or cask damage	Personnel injury Worker exposure to radioactive particulate Personnel contamination	Casks are sealed to prevent ingress of water. Fuel is dried before storing.	Casks are inspected. Casks are backfilled with inert gas.

Table D3-4. Safety Features for Facility Workers (S1 Consequence Items). (4 sheets)

Location/ checklist entry	Hazardous condition	Potential accident	Worker consequences	Features in addition to those identified in the design basis accident sections	
				Engineering features	Administrative features
Storage area/ J-11 M-01	Pyrophoric material exposed to air and/or water with sufficient surface area and temperature	Pyrophoric sodium metal reacts upon cladding breach	Personnel injury Contamination release	Structural design of cask (double seals) is such that breaches have a very low probability of occurring. Air ingress into cask would be limited by a small breach.	Casks are stored on isolated pad with limited access. Sodium is removed from exterior of FFTF fuel before storing.
Storage area/ L-01 L-02 L-03 L-07 L-14 L-16	Fire inside the ISA fence	Worker exposed to high heat from fire	Personnel injury	Support buildings and pad materials selected to minimize flammability. Casks can withstand design basis fires.	Flammable material inventories controlled within the fenced area.
Storage area/ P-01 P-03	External explosion resulting in a shock wave introduces dust, heat, and/or projectiles into the storage area. Radioactive material, toxic material, and direct radiation are present.	Shock wave impacts personnel Personnel are hit by debris Personnel are exposed to heat from explosion	Personnel injury Worker exposure to radioactive particulate	ISA is isolated from major roads, reducing the possibility of explosions outside the facility affecting the ISA. Distance of the ISA from other facilities reduces the consequences of external events.	Facility is addressed by the fire hazards analysis.

Table D3-4. Safety Features for Facility Workers (S1 Consequence Items). (4 sheets)

Location/ checklist entry	Hazardous condition	Potential accident	Worker consequences	Features in addition to those identified in the design basis accident sections	
				Engineering features	Administrative features
Storage area/ R-01 R-02 R-03 R-04 R-05 R-06 R-07 R-08 R-09 R-10	Forces exerted on the facility from lateral and horizontal accelerations ISA flooded Natural energy sources	Damage to or failure of structures, systems, or components ISA flooded	Personnel injury	ISA facility structure and confinement components are built to appropriate seismic criteria, as documented in the design. The physical location of the ISA and design of the storage pads and gravel areas result in flooding being incredible. ISA facility is designed for the lightning hazards. The facility structure is designed to withstand natural phenomena hazard loads.	Personnel are trained in sitewide and facility-specific emergency responses.

[a] TRIGA is a trademark of General Dynamics Corporation.
[b] NAC-1 casks are manufactured by Nuclear Assurance Corporation.
[c] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.
[d] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.

FFTF = Fast Flux Test Facility.
ISA = interim storage area.
ISC = interim storage cask.
LWR = light water reactor.
TSR = Technical Safety Requirement.

Table D3-5. Binned Listing of Candidates Sorted By Risk Ranking for 200 Area Interim Storage Area.

Candidate accident		Release energy	Risk ranking	Reference designator
D3.4.2.1	Handling/drop	low	5	G-03 G-06 G-13
D3.4.2.2	Mobile crane fall	low	5	G-03
D3.4.2.3	Cask tipover	low	5	G-03 G-06 G-13
D3.4.2.4	Fuel rod rupture	low	5	H-06 H-11
D3.4.2.5	Seismic	med/low	7	R-01
D3.4.2.6	Tornado/wind	med/low	7	R-06 R-08
D3.4.2.7	Fire	medium	5	L-03 L-07
D6.0	Inadvertent nuclear criticality	high	6	K-01 K-02 K-05 K-10 K-12 K-15

D3.3.3 Abnormal Events for the 200 Area Interim Storage Area Facility

Abnormal events are operating conditions resulting from situations outside of normal operations, where normal operations are defined by operation and maintenance procedures. These abnormal events encompass malfunctions of systems, operating upset conditions, or operator error. Abnormal events can be expected to occur annually or several times during the lifetime of the facility.

Abnormal events may impact operational or programmatic schedules; however, the consequences of the events are near zero or are standard industrial hazards that may include worker radiation exposure. Events having radiological consequences greater than allowed by the facility radiological protection and ALARA (as low as reasonably achievable) programs do not fit the abnormal event profile and are required to be analyzed as accidents by the DOE safety analysis process.

D3.4 ACCIDENT ANALYSIS

This section presents the methodology used to develop the DBAs identified in Section D3.3 and the results of the quantification of the consequences of those events. Also presented are the safety-class and safety-significant SSCs and TSRs related to these accident events that are necessary for protection of the offsite public and onsite workers. For each DBA, the following standard topics are discussed:

- Scenario development

- Source term analysis
- Consequence analysis
- Comparison to guidelines
- Summary of safety SSCs and TSRs.

D3.4.1 Methodology

This section identifies any ISA-specific methods, assumptions, or methodology used to quantify the consequences of the DBAs. The methods, assumptions, or methodology used to quantify the consequences of DBAs that are common or generic to all SNF Project facilities at the K Basins, Cold Vacuum Drying Facility, Canister Storage Building, and 200 Area ISA are described in Chapter 3.0 of HNF-3553, Volume 1.

D3.4.1.1 Source Term.

Hypothetical release source terms are developed for each fuel type in this section. The 200 Area ISA inventory assumed for the analysis is based on a per cask basis and either includes seven FFTF interim storage cask (ISC) assemblies, all 101 TRIGA elements, or one LWR assembly.

For this report, values for the "inhalation dose factor" were taken from Federal Guidance Report Number 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation* (EPA 1988). The computed committed dose for each radionuclide identified in the individual fuel types is calculated in Table D3-6, Table D3-7, and Table D3-8. To indicate the relative importance of the various radionuclides to the total dose, the last column shows the fraction contributed by each nuclide to the total dose per gram inhaled. As can be seen from the table, the radionuclides that account for a majority of the unit doses are ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am, and ^{244}Cm.

Note that the unit dose value presented in these tables assumes the uniform mixture of spent fuel is present in the airborne material. In other words, all nuclides have the same release fraction. The hypothetical accident scenarios have a preferential release of gaseous nuclides and require summing of the contributions of individual radionuclides.

The SNF is primarily uranium oxide, and mixed uranium and plutonium oxide, which are known to have toxicological effects. However, the toxicological consequences of the release of these substances will not require additional mitigating features beyond those already required by the radiological doses.

Table D3-6. Fast Flux Test Facility Radionuclide Composition of Fuel and the Unit Dose Factor per Assembly. (2 sheets)

Fission and Activation Products	Activity (Ci/Asmbly)	Activity (Bq/Asmbly)	Inhalation dose factor[a] (Sv/Bq)	Class	Unit dose[b] (Sv/Asmbly)	Percent of Total
H-3	7.82E+01	2.89E+12	2.60E-11[c]	Vapor	7.51E+01	0.00%
C-14	0.00E+00	0.00E+00	5.64E-10	Organic	0.00E+00	0.00%
Fe-55	0.00E+00	0.00E+00	7.26E-10	D	0.00E+00	0.00%
Co-60	0.00E+00	0.00E+00	8.94E-09	W	0.00E+00	0.00%
Ni-59	0.00E+00	0.00E+00	7.31E-10	Vapor	0.00E+00	0.00%
Ni-63	0.00E+00	0.00E+00	1.70E-09	Vapor	0.00E+00	0.00%
Se-79	0.00E+00	0.00E+00	2.66E-09	W	0.00E+00	0.00%
Kr-85	6.74E+02	2.49E+13	3.57E-13	Gas[d]	8.90E+00	0.00%
Sr-90	5.04E+03	1.86E+14	6.47E-08	D	1.21E+07	0.19%
Y-90	5.04E+03	1.86E+14	2.13E-09	W	3.97E+05	0.01%
Zr-93	0.00E+00	0.00E+00	8.67E-08	D	0.00E+00	0.00%
Nb-93m	0.00E+00	0.00E+00	8.68E-10	W	0.00E+00	0.00%
Tc-99	1.99E+00	7.36E+10	2.25E-09	W	1.66E+02	0.00%
Ru-106	4.89E+03	1.81E+14	3.18E-08	W	5.75E+06	0.09%
Rh-106	4.89E+03	1.81E+14	Daughter product[e]		0.00E+00	0.00%
Pd-107	0.00E+00	0.00E+00	2.19E-10	W	0.00E+00	0.00%
Ag-110	0.00E+00	0.00E+00	Daughter product[e]		0.00E+00	0.00%
Ag-110m	8.89E+00	3.29E+11	8.34E-09	W	2.74E+03	0.00%
Cd-113m	1.29E+01	4.77E+11	4.13E-07	D	1.97E+05	0.00%
In-113m	0.00E+00	0.00E+00	1.11E-11	D	0.00E+00	0.00%
Sn-113	0.00E+00	0.00E+00	2.88E-09	W	0.00E+00	0.00%
Sn-119m	0.00E+00	0.00E+00	1.69E-09	W	0.00E+00	0.00%
Sn-121m	0.00E+00	0.00E+00	3.11E-09	W	0.00E+00	0.00%
Sn-123	0.00E+00	0.00E+00	8.79E-09	W	0.00E+00	0.00%
Sn-126	0.00E+00	0.00E+00	2.69E-08	W	0.00E+00	0.00%
Sb-125	1.21E+03	4.48E+13	3.30E-09	W	1.48E+05	0.00%
Sb-126	0.00E+00	0.00E+00	3.17E-09	W	0.00E+00	0.00%
Sb-126m	0.00E+00	0.00E+00	9.17E-12	D	0.00E+00	0.00%
Te-123m	0.00E+00	0.00E+00	2.86E-09	W	0.00E+00	0.00%
Te-125m	2.96E+02	1.10E+13	1.97E-09	W	2.16E+04	0.00%
Te-127	0.00E+00	0.00E+00	8.60E-11	W	0.00E+00	0.00%
Te-127m	0.00E+00	0.00E+00	5.81E-09	W	0.00E+00	0.00%
I-129	6.63E-03	2.45E+08	4.69E-08	D	1.15E+01	0.00%
Cs-134	2.51E+03	9.29E+13	1.25E-08	D	1.16E+06	0.02%
Cs-135	0.00E+00	0.00E+00	1.23E-09	D	0.00E+00	0.00%
Cs-137	1.37E+04	5.07E+14	8.63E-09	D	4.37E+06	0.07%
Ba-137m	1.30E+04	4.80E+14	Daughter product[e]		0.00E+00	0.00%
Ce-144	1.66E+03	6.14E+13	5.84E-08	W	3.59E+06	0.06%

Table D3-6. Fast Flux Test Facility Radionuclide Composition of Fuel and the Unit Dose Factor per Assembly. (2 sheets)

Fission and Activation Products	Activity (Ci/Asmbly)	Activity (Bq/Asmbly)	Inhalation dose factor[a] (Sv/Bq)	Class	Unit dose[b] (Sv/Asmbly)	Percent of Total
Pr-144	1.66E+03	6.14E+13	1.10E-11	W	6.76E+02	0.00%
Pr-144m	2.37E+01	8.78E+11	Daughter product[e]		0.00E+00	0.00%
Pm-147	8.73E+03	3.23E+14	6.97E-09	W	2.25E+06	0.04%
Sm-151	5.21E+02	1.93E+13	8.10E-09	W	1.56E+05	0.00%
Eu-152	0.00E+00	0.00E+00	5.97E-08	W	0.00E+00	0.00%
Eu-154	0.00E+00	0.00E+00	7.73E-08	W	0.00E+00	0.00%
Eu-155	0.00E+00	0.00E+00	1.12E-08	W	0.00E+00	0.00%
Gd-153	0.00E+00	0.00E+00	6.43E-09	D	0.00E+00	0.00%
Subtotal:	6.39E+04	2.36E+15			3.01E+07	0.48%
			Actinides			
U-234	0.00E+00	0.00E+00	2.13E-06	W	0.00E+00	0.00%
U-235	0.00E+00	0.00E+00	1.97E-06	W	0.00E+00	0.00%
U-236	0.00E+00	0.00E+00	2.01E-06	W	0.00E+00	0.00%
U-238	0.00E+00	0.00E+00	1.91E-06	W	0.00E+00	0.00%
Np-237	0.00E+00	0.00E+00	1.46E-04	W	0.00E+00	0.00%
Pu-238	2.81E+02	1.04E+13	1.06E-04	W	1.10E+09	17.66%
Pu-239	3.49E+02	1.29E+13	1.16E-04	W	1.50E+09	24.00%
Pu-240	2.68E+02	9.92E+12	1.16E-04	W	1.15E+09	18.43%
Pu-241	1.46E+04	5.40E+14	2.23E-06	W	1.20E+09	19.30%
Pu-242	0.00E+00	0.00E+00	1.11E-04	W	0.00E+00	0.00%
Am-241	2.53E+02	9.36E+12	1.20E-04	W	1.12E+09	18.00%
Am-242	1.92E+01	7.11E+11	1.58E-08	W	1.12E+04	0.00%
Am-242m	1.93E+01	7.14E+11	1.15E-04	W	8.21E+07	1.32%
Am-243	0.00E+00	0.00E+00	1.19E-04	W	0.00E+00	0.00%
Cm-242	0.00E+00	0.00E+00	4.67E-06	W	0.00E+00	0.00%
Cm-244	2.04E+01	7.55E+11	6.70E-05	W	5.06E+07	0.81%
Subtotal:	1.58E+04	5.85E+14			6.21E+09	99.52%
				Total:	6.24E+09	(Sv/Asmbly)

[a] Inhalation dose factors from Federal Guidance Report Number 11, 1988, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation*, U.S. Environmental Protection Agency, Washington, D.C.

[b] The unit dose is the product of the normalized activity and the inhalation dose factor. To convert Sv to rem, multiply by 100.

[c] Internal dose factor for tritium was increased by 50% to include skin absorption.

[d] Kr-85 is a noble gas. It does not accumulate in the body; therefore, its internal dose factor is zero. The value shown is the external dose rate factor for submersion in an infinite cloud divided by the light activity breathing rate.

[e] Daughter products are included with parents and not tracked individually.

Table D3-7. Neutron Radiography Facility TRIGA Radionuclide Composition of Fuel and the Unit Dose Factor per Assembly. (2 sheets)

Fission and Activation Products	Activity (Ci/Asmbly)	Activity (Bq/Asmbly)	Inhalation dose factor[a] (Sv/Bq)	Class	Unit dose[b] (Sv/Asmbly)	Percent of Total
H-3	5.16E-03	1.91E+08	2.60E-11[c]	Vapor	4.95E-03	0.00%
C-14	1.21E-05	4.47E+05	5.64E-10	Organic	2.52E-04	0.00%
Fe-55	1.90E-01	7.04E+09	7.26E-10	D	5.11E+00	0.02%
Co-60	6.46E-04	2.39E+07	8.94E-09	W	2.14E-01	0.00%
Ni-59	5.49E-04	2.03E+07	7.31E-10	Vapor	1.48E-02	0.00%
Ni-63	5.94E-02	2.20E+09	1.70E-09	Vapor	3.74E+00	0.01%
Se-79	0.00E+00	0.00E+00	2.66E-09	W	0.00E+00	0.00%
Kr-85	1.27E-01	4.71E+09	3.57E-13	Gas [d]	1.68E-03	0.00%
Sr-90	1.86E+00	6.90E+10	6.47E-08	D	4.46E+03	15.84%
Y-90	1.86E+00	6.90E+10	2.13E-09	W	1.47E+02	0.52%
Zr-93	2.65E-05	9.80E+05	8.67E-08	D	8.49E-02	0.00%
Nb-93m	0.00E+00	0.00E+00	8.68E-10	W	0.00E+00	0.00%
Tc-99	4.06E-04	1.50E+07	2.25E-09	W	3.38E-02	0.00%
Ru-106	3.39E-02	1.25E+09	3.18E-08	W	3.99E+01	0.14%
Rh-106	3.39E-02	1.25E+09	Daughter product[e]		0.00E+00	0.00%
Pd-107	0.00E+00	0.00E+00	2.19E-10	W	0.00E+00	0.00%
Ag-110	0.00E+00	0.00E+00	Daughter product[e]		0.00E+00	0.00%
Ag-110m	0.00E+00	0.00E+00	8.34E-09	W	0.00E+00	0.00%
Cd-113m	1.94E-04	7.19E+06	4.13E-07	D	2.97E+00	0.01%
In-113m	0.00E+00	0.00E+00	1.11E-11	D	0.00E+00	0.00%
Sn-113	0.00E+00	0.00E+00	2.88E-09	W	0.00E+00	0.00%
Sn-119m	0.00E+00	0.00E+00	1.69E-09	W	0.00E+00	0.00%
Sn-121m	0.00E+00	0.00E+00	3.11E-09	W	0.00E+00	0.00%
Sn-123	0.00E+00	0.00E+00	8.79E-09	W	0.00E+00	0.00%
Sn-126	0.00E+00	0.00E+00	2.69E-08	W	0.00E+00	0.00%
Sb-125	1.37E-02	5.06E+08	3.30E-09	W	1.67E+00	0.01%
Sb-126	0.00E+00	0.00E+00	3.17E-09	W	0.00E+00	0.00%
Sb-126m	0.00E+00	0.00E+00	9.17E-12	D	0.00E+00	0.00%
Te-123m	0.00E+00	0.00E+00	2.86E-09	W	0.00E+00	0.00%
Te-125m	3.34E-03	1.23E+08	1.97E-09	W	2.43E-01	0.00%
Te-127	0.00E+00	0.00E+00	8.60E-11	W	0.00E+00	0.00%
Te-127m	0.00E+00	0.00E+00	5.81E-09	W	0.00E+00	0.00%
I-129	0.00E+00	0.00E+00	4.69E-08	D	0.00E+00	0.00%
Cs-134	9.10E-03	3.37E+08	1.25E-08	D	4.21E+00	0.01%
Cs-135	0.00E+00	0.00E+00	1.23E-09	D	0.00E+00	0.00%
Cs-137	1.97E+00	7.27E+10	8.63E-09	D	6.27E+02	2.23%
Ba-137m	1.86E+00	6.88E+10	Daughter product[e]		0.00E+00	0.00%

Table D3-7. Neutron Radiography Facility TRIGA Radionuclide Composition of Fuel and the Unit Dose Factor per Assembly. (2 sheets)

Fission and Activation Products	Activity (Ci/Asmbly)	Activity (Bq/Asmbly)	Inhalation dose factor[a] (Sv/Bq)	Class	Unit dose[b] (Sv/Asmbly)	Percent of Total
Ce-144	1.86E-01	6.89E+09	5.84E-08	W	4.03E+02	1.43%
Pr-144	1.86E-01	6.89E+09	1.10E-11	W	7.58E-02	0.00%
Pr-144m	2.66E-03	9.86E+07	Daughter product[e]		0.00E+00	0.00%
Pm-147	6.73E-01	2.49E+10	6.97E-09	W	1.74E+02	0.62%
Sm-151	4.99E-02	1.85E+09	8.10E-09	W	1.50E+01	0.05%
Eu-152	2.76E-03	1.02E+08	5.97E-08	W	6.09E+00	0.02%
Eu-154	2.11E-03	7.82E+07	7.73E-08	W	6.05E+00	0.02%
Eu-155	1.51E-02	5.57E+08	1.12E-08	W	6.24E+00	0.02%
Gd-153	0.00E+00	0.00E+00	6.43E-09	D	0.00E+00	0.00%
Subtotal:	9.15E+00	3.38E+11			5.90E+03	20.96%
			Actinides			
U-234	2.35E-03	8.70E+07	2.13E-06	W	1.85E+02	0.66%
U-235	7.97E-05	2.95E+06	1.97E-06	W	5.81E+00	0.02%
U-236	0.00E+00	0.00E+00	2.01E-06	W	0.00E+00	0.00%
U-238	5.17E-05	1.91E+06	1.91E-06	W	3.66E+00	0.01%
Np-237	1.77E-06	6.55E+04	1.46E-04	W	9.56E+00	0.03%
Pu-238	3.16E-04	1.17E+07	1.06E-04	W	1.24E+03	4.39%
Pu-239	4.46E-03	1.65E+08	1.16E-04	W	1.91E+04	67.87%
Pu-240	2.87E-04	1.06E+07	1.16E-04	W	1.23E+03	4.37%
Pu-241	2.67E-03	9.88E+07	2.23E-06	W	2.20E+02	0.78%
Pu-242	0.00E+00	0.00E+00	1.11E-04	W	0.00E+00	0.00%
Am-241	5.75E-05	2.13E+06	1.20E-04	W	2.55E+02	0.91%
Am-242	0.00E+00	0.00E+00	1.58E-08	W	0.00E+00	0.00%
Am-242m	0.00E+00	0.00E+00	1.15E-04	W	0.00E+00	0.00%
Am-243	0.00E+00	0.00E+00	1.19E-04	W	0.00E+00	0.00%
Cm-242	0.00E+00	0.00E+00	4.67E-06	W	0.00E+00	0.00%
Cm-244	0.00E+00	0.00E+00	6.70E-05	W	0.00E+00	0.00%
Subtotal:	1.03E-02	3.80E+08			2.23E+04	79.04%
				Total:	2.82E+04	(Sv/Asmbly)

Note: TRIGA is a trademark of General Dynamics Corporation.

[a] Inhalation dose factors from Federal Guidance Report Number 11, 1988, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation*, U.S. Environmental Protection Agency, Washington, D.C.

[b] The unit dose is the product of the normalized activity and the inhalation dose factor. To convert Sv to rem, multiply by 100.

[c] Internal dose factor for tritium was increased by 50% to include skin absorption.

[d] Kr-85 is a noble gas. It does not accumulate in the body; therefore, its internal dose factor is zero. The value shown is the external dose rate factor for submersion in an infinite cloud divided by the light activity breathing rate.

[e] Daughter products are included with parents and not tracked individually.

Table D3-8. Light Water Reactor Radionuclide Composition of Fuel and the Unit Dose Factor per Assembly. (2 sheets)

Fission and Activation Products	Activity (Ci/Asmbly)	Activity (Bq/Asmbly)	Inhalation dose factor[a] (Sv/Bq)	Class	Unit dose[b] (Sv/Asmbly)	Percent of Total
H-3	9.28E+01	3.43E+12	2.60E-11[c]	Vapor	8.91E+01	0.00%
C-14	0.00E+00	0.00E+00	5.64E-10	Organic	0.00E+00	0.00%
Fe-55	0.00E+00	0.00E+00	7.26E-10	D	0.00E+00	0.00%
Co-60	7.88E+02	2.92E+13	8.94E-09	W	2.61E+05	0.00%
Ni-59	0.00E+00	0.00E+00	7.31E-10	Vapor	0.00E+00	0.00%
Ni-63	0.00E+00	0.00E+00	1.70E-09	Vapor	0.00E+00	0.00%
Se-79	0.00E+00	0.00E+00	2.66E-09	W	0.00E+00	0.00%
Kr-85	1.31E+03	4.85E+13	3.57E-13	Gas[d]	1.73E+01	0.00%
Sr-90	1.98E+04	7.33E+14	6.47E-08	D	4.74E+07	0.29%
Y-90	1.98E+04	7.33E+14	2.13E-09	W	1.56E+06	0.01%
Zr-93	0.00E+00	0.00E+00	8.67E-08	D	0.00E+00	0.00%
Nb-93m	0.00E+00	0.00E+00	8.68E-10	W	0.00E+00	0.00%
Tc-99	5.21E+00	1.93E+11	2.25E-09	W	4.34E+02	0.00%
Ru-106	8.43E+00	3.12E+11	3.18E-08	W	9.92E+03	0.00%
Rh-106	8.43E+00	3.12E+11	Daughter product[e]		0.00E+00	0.00%
Pd-107	0.00E+00	0.00E+00	2.19E-10	W	0.00E+00	0.00%
Ag-110	0.00E+00	0.00E+00	Daughter product[e]		0.00E+00	0.00%
Ag-110m	0.00E+00	0.00E+00	8.34E-09	W	0.00E+00	0.00%
Cd-113m	1.31E+01	4.85E+11	4.13E-07	D	2.00E+05	0.00%
In-113m	0.00E+00	0.00E+00	1.11E-11	D	0.00E+00	0.00%
Sn-113	0.00E+00	0.00E+00	2.88E-09	W	0.00E+00	0.00%
Sn-119m	0.00E+00	0.00E+00	1.69E-09	W	0.00E+00	0.00%
Sn-121m	0.00E+00	0.00E+00	3.11E-09	W	0.00E+00	0.00%
Sn-123	0.00E+00	0.00E+00	8.79E-09	W	0.00E+00	0.00%
Sn-126	0.00E+00	0.00E+00	2.69E-08	W	0.00E+00	0.00%
Sb-125	1.30E+02	4.81E+12	3.30E-09	W	1.59E+04	0.00%
Sb-126	0.00E+00	0.00E+00	3.17E-09	W	0.00E+00	0.00%
Sb-126m	0.00E+00	0.00E+00	9.17E-12	D	0.00E+00	0.00%
Te-123m	0.00E+00	0.00E+00	2.86E-09	W	0.00E+00	0.00%
Te-125m	3.17E+01	1.17E+12	1.97E-09	W	2.31E+03	0.00%
Te-127	0.00E+00	0.00E+00	8.60E-11	W	0.00E+00	0.00%
Te-127m	0.00E+00	0.00E+00	5.81E-09	W	0.00E+00	0.00%
I-129	1.26E-02	4.66E+08	4.69E-08	D	2.19E+01	0.00%
Cs-134	4.91E+02	1.82E+13	1.25E-08	D	2.27E+05	0.00%
Cs-135	0.00E+00	0.00E+00	1.23E-09	D	0.00E+00	0.00%
Cs-137	2.85E+04	1.05E+15	8.63E-09	D	9.10E+06	0.06%

Table D3-8. Light Water Reactor Radionuclide Composition of Fuel and the Unit Dose Factor per Assembly. (2 sheets)

Fission and Activation Products	Activity (Ci/Asmbly)	Activity (Bq/Asmbly)	Inhalation dose factor[a] (Sv/Bq)	Class	Unit dose[b] (Sv/Asmbly)	Percent of Total
Ba-137m	2.70E+04	9.98E+14	Daughter product[e]		0.00E+00	0.00%
Ce-144	0.00E+00	0.00E+00	5.84E-08	W	0.00E+00	0.00%
Pr-144	0.00E+00	0.00E+00	1.10E-11	W	0.00E+00	0.00%
Pr-144m	0.00E+00	0.00E+00	Daughter product[e]		0.00E+00	0.00%
Pm-147	8.17E+02	3.02E+13	6.97E-09	W	2.11E+05	0.00%
Sm-151	1.23E+02	4.55E+12	8.10E-09	W	3.69E+04	0.00%
Eu-152	0.00E+00	0.00E+00	5.97E-08	W	0.00E+00	0.00%
Eu-154	1.55E+03	5.74E+13	7.73E-08	W	4.43E+06	0.03%
Eu-155	4.02E+02	1.49E+13	1.12E-08	W	1.67E+05	0.00%
Gd-153	0.00E+00	0.00E+00	6.43E-09	D	0.00E+00	0.00%
Subtotal:	1.01E+05	3.73E+15			6.36E+07	0.39%
			Actinides			
U-234	4.72E-01	1.75E+10	2.13E-06	W	3.72E+04	0.00%
U-235	6.91E-03	2.56E+08	1.97E-06	W	5.04E+02	0.00%
U-236	1.07E-01	3.96E+09	2.01E-06	W	7.96E+03	0.00%
U-238	1.30E-01	4.81E+09	1.91E-06	W	9.19E+03	0.00%
Np-237	1.34E-01	4.96E+09	1.46E-04	W	7.24E+05	0.00%
Pu-238	1.36E+03	5.03E+13	1.06E-04	W	5.33E+09	33.01%
Pu-239	1.25E+02	4.63E+12	1.16E-04	W	5.37E+08	3.32%
Pu-240	2.12E+02	7.84E+12	1.16E-04	W	9.10E+08	5.63%
Pu-241	2.44E+04	9.03E+14	2.23E-06	W	2.01E+09	12.46%
Pu-242	9.39E-01	3.47E+10	1.11E-04	W	3.86E+06	0.02%
Am-241	9.44E+02	3.49E+13	1.20E-04	W	4.19E+09	25.94%
Am-242	1.06E+01	3.94E+11	1.58E-08	W	6.23E+03	0.00%
Am-242m	1.07E+01	3.96E+11	1.15E-04	W	4.55E+07	0.28%
Am-243	1.28E+01	4.74E+11	1.19E-04	W	5.64E+07	0.35%
Cm-242	8.82E+00	3.26E+11	4.67E-06	W	1.52E+06	0.01%
Cm-244	1.21E+03	4.48E+13	6.70E-05	W	3.00E+09	18.57%
Subtotal:	2.83E+04	1.05E+15			1.61E+10	99.61%
				Total:	1.62E+10	(Sv/Asmbly)

[a] Inhalation dose factors from Federal Guidance Report Number 11, 1988, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation*, U.S. Environmental Protection Agency, Washington, D.C.

[b] The unit dose is the product of the normalized activity and the inhalation dose factor. To convert Sv to rem, multiply by 100.

[c] Internal dose factor for tritium was increased by 50% to include skin absorption.

[d] Kr-85 is a noble gas. It does not accumulate in the body; therefore, its internal dose factor is zero. The value shown is the external dose rate factor for submersion in an infinite cloud divided by the light activity breathing rate.

[e] Daughter products are included with parents and not tracked individually.

D3.4.1.2 Consequence Analysis.

For each SNF type, there is a bounding credible accident for which a maximum credible radiological release and dose consequence is determined. These radiological doses to a maximum duration exposed onsite and offsite receptor are estimated by using the following equation:

$$D = \frac{\chi}{Q'} \times BR \times (UD \times RF_{total} \times \#_{assemblies})$$

where

D	=	committed effective dose equivalent (Sv)
χ/Q'	=	atmospheric transport factor (s/m^3)
BR	=	breathing rate (m^3/s)
UD	=	50-year committed dose per unit respirable radioactive material inhaled (Sv/assembly])
RF_{total}	=	combined release fraction for each actinide
$\#_{assemblies}$	=	number of assemblies (since UD is developed as SV/assembly).

The quantity of respirable material released is dependent on the specific release fractions developed in this section. The total Sv released equals the UD times the release fractions for each fuel type times the number of assemblies involved. The number of fuel assemblies is established in Section D3.4.1.1. The parameters and values of χ/Q' and breathing rate (BR) are defined and specified in their respective analyses. The accident-specific, location-specific, release fraction and fuel type-specific onsite and offsite dose consequences are estimated from this formulation.

The atmospheric transport factor (χ/Q') is based on specific release conditions (e.g., ground level or elevated, long or short duration) and the receptor's distance from the release. While the methodology is common to the SNF Project, the atmospheric transport factor is the time-integrated normalized air concentration at the receptor's location, which is a measured distance from the Canister Storage Building. The transport factor includes the dilution of an airborne contaminant caused by atmospheric mixing and turbulence. The air transport values used in this report have been generated using the GXQ computer program, which is documented in WHC-SD-GN-SWD-30002, *GXQ 4.0 Program Users' Guide*, and WHC-SD-GN-SWD-30003, *GXQ 4.0 Program Verification & Validation.* Table D3-9 contains the air transport values used to determine onsite and offsite consequences.

Air transport factors were calculated using methods found in NRC Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*. In each wind direction, the observed frequencies of particular wind speed and stability class combinations are used to compute a value that is exceeded only 0.5% of the time. This is repeated for all 16 compass directions to determine the worst-case location.

Table D3-9. Atmospheric Transport Factors Used in Accident Analyses for the Interim Storage Area.

Receptor location description	Air transport factors[a] Acute Less than 1 hour[b]
Onsite worker	3.41 E-02
Highway 240	2.36 E-05
Hanford Site boundary	1.30 E-05

[a] Units for these values are seconds per cubic meter. In all cases, the releases are assumed to be point sources at ground level to maximize the dose consequences.
[b] No adjustment for plume meander (HNF-SD-SNF-TI-059, 1999, *A Discussion on the Methodology for Calculating Radiological and Toxicological Consequences for the Spent Nuclear Fuel Project at the Hanford Site*, Rev. 2, Fluor Daniel Northwest, Incorporated, Richland, Washington).

Exposures to the collocated worker onsite are calculated for the individual at the 100-m location. The onsite risk evaluation guidelines apply to this individual. For assessment purposes, DOE Letter 9601270B/96-SFD-113, *Clarification of Site Boundary for Spent Nuclear Fuel Project (SNFP) Work in or the Near K Basins* (Sellers 1996), directed that the Hanford Site boundary be considered the location of the offsite receptor. Highway 240 is not used as the nearest public access because DOE and its contractors can control access during emergency and accident conditions. This access control meets the requirements of 10 CFR 72.106(c).

None of the accidents analyzed in this document adjusts the air transport factors for the finite size of the source (i.e., cask wake effects) or for the elevation of the release above ground level (i.e., stack effects). Section 1.4.1 of HNF-3553, Volume 1, provides additional information on the calculation of the air transport factors. The basis for defining the location of the onsite and offsite receptors is provided in Section D1.3.1.3.

The breathing rate (BR) depends on individual activity factors and exposure duration. This methodology is common to the SNF Project facilities and is described in Chapter 3.0 of HNF-3553, Volume 1.

The dose per unit intake (UD) is the 50-year dose commitment for all relevant exposure pathways per gram of radioactive material inhaled. The major radiation exposure pathway for the identified accidents is inhalation of radioactive material. This methodology is common to the SNF Project facilities and is described in Chapter 3.0 of HNF-3553, Volume 1.

Hypothetical Release Fractions

An unmitigated release from each fuel type cask is evaluated to determine the appropriate safety classification of the engineered barriers that prevent an uncontrolled release. The hypothetical accident scenario developed in this analysis assumes a major failure of the cask with crushing of some fuel pellets. Some credit is taken for the physical presence of the cask around the fuel, even though its design confinement integrity has been breached. The hypothetical unmitigated release is developed based on the quantity of radioactive material that might be released if the fuel assembly or assemblies within the storage cask are damaged when the cask is breached. Although this scenario is hypothetical and non-mechanistic, the scenario establishes the magnitude of a potential release. Based on the postulated consequences of the unmitigated release, appropriate safety classification of the protection SSCs can then be accomplished.

For hypothetical purposes, a scenario is proposed involving a crane hook structure, with a weight of 2 tons falling from a height of 80 ft (but hypothetically retaining significant energy to crush the fuel after puncturing the cask sidewall). The tapered shape of the hook bottom is assumed to be about 2 in. across the leading edge, tapering out to about 5 in. thick at the throat of the hook. This hook creates a strong rigid projectile that (for this scenario) is hypothetically proposed to have enough energy to penetrate the cask sidewalls and shear all fuel rods contained within the cask and inner canister. This hypothetical accident results in clad damage to all of the elements and pulverization of some of the fuel that was directly struck by the crane hook. This accident would conservatively damage about a 7-in. long cross-section of the fuel assembly. In this scenario, it only has sufficient energy remaining to crush half of the material damaged, which equates to about 2.5% of the fuel inventory in the cask. The effective impact energy density calculated for the hook structure falling from a height of 80 ft is 1.7×10^9 ergs/cm^3. Figure 4.11 of NUREG-1320, *Nuclear Fuel Cycle Facility Accident Analysis Hand Book*, indicates that the mass fraction of particulate material less than 10 μm generated from this energy density is about 1.0% for ceramic fuel. Figure 4.15 (NUREG-1320) indicates that about 10% of the less than 10 μm particles are less than 3 μm, which is considered respirable for a particle with a density like uranium oxide. Credit is taken for the damaged cask structure retaining 90% of the particulate generated.

The parameters relating to fuel damage and release fractions for the postulated scenarios are listed below and in Table D3-10.

Fraction of fuel pins damaged	-	2.5%
Gap noble gases, iodine, tritium	-	100% released from damaged pins
Gap particulates released	-	1% released from damaged pins
Particulates from crushed pins	-	1%
Respirable fraction of particulates	-	10%
Fraction of noble gases, iodine, tritium released from crushed pins	-	100%
Fraction of noble gases, iodine, tritium released from cask	-	100%
Fraction of particulates released from casks	-	10%

Table D3-10. Dose Calculation Summary for the Postulated Releases.

Receptor identity	Air transport factor, s/m^3	Sv released	Breathing Rate, m^3/s[a]	CEDE, rem[b]	Anticipated evaluation guidelines[c]
Fast Flux Test Facility Fuel (7 assemblies)					
Onsite worker, 100 m east-southeast	3.41E-02	1.10E+05	3.33E-04	1.25E+02	1
Highway 240, 9,280 m west	2.36E-05	1.10E+05	3.33E-04	8.63E-02	0.5
Hanford boundary, 17,390 m east	1.30E-05	1.10E+05	3.33E-04	4.75E-02	0.5
Neutron Radiography Facility TRIGA[d] Fuel (101 assemblies)					
Onsite worker, 100 m east-southeast	3.41E-02	7.93E0	3.33E-04	9.01E-03	1
Highway 240, 9,280 m west	2.36E-05	7.93E0	3.33E-04	6.24E-06	0.5
Hanford boundary, 17,390 m east	1.30E-05	7.93E0	3.33E-04	3.44E-06	0.5
Commercial Light Water Reactor Fuel (1 assembly)					
Onsite worker, 100 m east-southeast	3.41E-02	4.05E+04	3.33E-04	4.60E+01	1
Highway 240, 9,280 m west	2.36E-05	4.05E+04	3.33E-04	3.19E-02	0.5
Hanford boundary, 17,390 m east	1.30E-05	4.05E+04	3.33E-04	1.76E-02	0.5

[a] The average breathing rate is: light activity breathing rate.
[b] To convert Sv to rem, multiply by 100.
[c] Guideline is for the "anticipated" accident category.
[d] TRIGA is a trademark of General Dynamics Corporation.

CEDE = committed effective dose equivalent.

The release is modeled to enter the environment as a ground level airborne release with a 1-hour duration. Radiological doses resulting from each of the postulated accidents were calculated for the maximum onsite worker and offsite individual. The analysis shown in the tables indicates that even for the extremely conservative postulated accident of an unmitigated release, radiation exposure to maximum offsite individuals would be less than 0.5 rem.

D3.4.1.3 Frequency Estimates.

DOE Order 6430.1A requires that safety-class SSCs be identified if they are required to place or maintain an operating process in a safe condition when that process prevents or mitigates consequences to the public greater than 500 mrem (5 mSv) total effective dose equivalent, independent of the estimated event frequency. Therefore, the determination of frequency estimates is based on methodology common to the SNF Project facilities and is described in Chapter 3.0 of HNF-3553, Volume 1.

D3.4.1.4 Risk Guidelines.

The DOE-recommended radiological risk evaluation guidelines (Sellers 1997) are applied across the SNF Project and are described in Chapter 3.0 of HNF-3553, Volume 1.

D3.4.1.5 Safety Structures, Systems, and Components.

Safety-class and safety-significant designations, as related to the SSCs, are defined consistently for the SNF Project in Chapter 3.0 of HNF-3553, Volume 1.

D3.4.2 Design Basis Accident Analysis

Although the 200 Area ISA is a new facility, the development of safety documentation for a new project per HNF-PRO-703, *Safety Analysis Process - New Project*, is different for this FSAR. The safety bases for the FFTF SNF and the TRIGA SNF storage systems and the 400 Area ISA site, as documented in WHC-TI-75002, *Fast Flux Test Facility Final Safety Analysis Report, Amendment 80*, Appendix H, "Fast Flux Test Facility Fuel Offload and Fuel Storage in the Interim Storage Area," were approved under a DOE Richland Operations Office-approved safety authorization basis at FFTF (Wagoner 1996). These storage systems and SNF types have been analyzed and approved for storage at the 400 Area ISA. Although these analyses were performed for the 400 Area ISA, information relevant to the performance of the FFTF and TRIGA storage systems at the 200 Area ISA can be derived from many of the analyses.

The existing analyses for the FFTF, TRIGA, and LWR spent fuel were evaluated for their applicability to the 200 Area ISA. The evaluations reviewed the assumptions, methodology, parameters, parameter values, results, and conclusions of the 400 Area ISA analyses for application to the 200 Area ISA. The results of these evaluations are provided in this section. The evaluations generally found that the analyses that relied solely on the performance and functional characteristics of the storage systems could be applied from the 400 Area ISA to the 200 Area ISA, but where site characteristics of the 400 Area ISA were not enveloped by the site characteristics of the 200 Area ISA, new analyses were performed.

The results of these evaluations and the 200 Area ISA site-specific analyses provide the bases to: (1) identify the hazards, energy sources, potential accident sequences, available mitigating barriers and controls, and qualitative estimates of accident frequency and consequence; (2) evaluate the most significant accident(s) to provide the technical bases for the conclusions related to safety, health, and the environment (e.g., final safety designation and comparison with risk guidelines); and (3) provide a final assessment of the design relative to maintaining its confinement function under normal, abnormal, and accident conditions.

This section presents a summary of the key assumptions and results of the DBA analyses that have been performed for the ISA. The DBAs are summarized based on the guidelines provided in DOE-STD-3009-94, and include the following categories:

- Cask handling/drop
- Mobile crane fall
- Cask tipover
- Fuel rod rupture
- Seismic
- Tornado/wind
- Fire.

The DBAs have been analyzed to quantify consequences and compare them with release limits for offsite consequences and evaluation guidelines for onsite consequences. The process is iterative, starting by taking no credit for mitigative features and comparing the results to the limits or guidelines. Credit is then taken for safety SSCs that prevent or mitigate the consequences to show that the results are below the release limits and evaluation guidelines. The process continues after the release limits and evaluation guidelines are met by identifying other SSCs that, while not designated as safety class or safety significant, provide additional mitigative features as defense in depth.

D3.4.2.1 Handling and Drop Accidents.

D3.4.2.1.1 Scenario Development.

While handling the storage casks, an accidental drop or other handling impact could possibly occur. If the drop or impact caused a cask to undergo accelerations and forces beyond the design strength, the cask confinement could be breached. Breach of the cask confinement could result in radiological releases. In addition, as a passive design feature, each FFTF ISC and NAC-1[3] cask protects the geometry of the stored SNF during and after impact to assure criticality safety. The handling and drop scenarios are developed in this section for each fuel type. A summary of the safety features required to prevent handling and drop accidents is provided in Table D3-11.

[3] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

Table D3-11. Summary of Safety Features Required to Prevent Handling/Drop Accident.

Candidate accident	Checklist designator[a]	Safety function	Safety features (described in Chapter D4.0)
Drop of the cask while being moved by a crane	G-03 G-06 G-13	Maintain geometry control after credible drops. Maintain confinement of radioactive materials after a credible drop and provide passive protection such that structural integrity is maintained.	Safety class equipment for criticality geometry control: • CCC • LWR canister. Safety-significant equipment for confinement: • FFTF ISC • LWR canister. Safety-significant equipment for structural integrity: • FFTF ISC • NAC-1[b] cask. TSR: • Restriction on cask lift heights. • Restriction on lifting casks over objects. Defense-in-depth: • NRF TRIGA[c] cask • DOT-6M[d] container • 2R container[e] • Qualified crane operators • Detailed procedures.

[a] Checklist designators are from SNF-4820, 1999, *200 Area Interim Storage Area Final Hazard Analysis Report*, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.
[b] NAC-1 casks are manufactured by Nuclear Assurance Corporation.
[c] TRIGA is a trademark of General Dynamics Corporation.
[d] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.
[e] 2R designates the inner container of DOT-6M.

CCC = core component container.
FFTF = Fast Flux Test Facility.
ISC = interim storage cask.
LWR = light water reactor.
NRF = Neutron Radiography Facility.
TSR = Technical Safety Requirement.

Fast Flux Test Facility Fuel

The FFTF fuel package consists of fuel assemblies or pin containers contained within a core component container (CCC), which in turn is contained within an ISC. One of the CCC drop analyses consists of a postulated drop into the ISC. Although the scope of this analysis does not include CCC removal from or placement in the ISC, the analysis provides a measure of the impact strength of the CCC to ensure that the CCC can remain intact inside the ISC for geometry (criticality) control. The ISC, with its contents, has also been analyzed to ensure that confinement of the fuel is maintained.

The CCC was analyzed in WHC-SD-FF-DA-077, *Stress and Structural Analysis of the Core Component Container*, to confirm that it could safely withstand accident conditions. Fully loaded CCCs (six or seven fuel assemblies, or six pin containers) are addressed in WHC-SD-FF-DA-077 and SNF-4790, *200 Area Interim Storage Design Basis Accident Analysis Documentation for FFTF Fuel Storage*. Partially loaded CCCs are not analyzed, and must be evaluated on a case-by-case basis. Material properties were evaluated using a peak CCC temperature condition of 600 °F. Normal CCC conditions were evaluated in accordance with stress limits of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code*, Section VIII, Division 2 (ASME 1989), and the accident conditions were evaluated in accordance with stress limits of the ASME Code, Section III, Appendix F (ASME 1989).

The accident conditions analyzed for the CCC are as follows:

- 18-ft drop of the CCC into the ISC
- 4-ft sideways drop of the CCC while inside the ISC.

Several drop analyses have been performed for the ISC. Under normal operations, the ISC is lifted vertically by attaching the lifting fixture to three anchor attachments imbedded in the concrete of the ISC. No side lifting is allowed. In accordance with the critical lift requirements of DOE/RL-92-36, *Hanford Site Hoisting and Rigging Manual*, the ISC lift points are designed to lift five times the weight of the cask, without exceeding the ultimate stress of the material, or three times the weight, without exceeding the yield strength of the material, whichever is less. Therefore, the normal ISC lifting and handling loads generate low stresses.

The weakest link in the ISC attachment design is the anchor lug, which screws into the anchor bolt of the cask. If a lifting accident occurs whereby a lift point fails, this anchor will fail before the ISC concrete embedment. Thus, the ISC integrity or shielding will not be affected.

The accident conditions analyzed for the ISC are as follows:

- 18-ft drop of the CCC into the ISC
- 4-ft sideways drop of the ISC
- 4-ft ISC drop onto an unyielding surface
- 8-ft ISC drop with the CCC inside onto a 1.5-ft thick concrete pad
- ISC drop onto another object
- 40-in. ISC drop onto a 6-in. diameter mild steel punch.

18-ft drop of the CCC into the ISC. The ISC internal impact limiter is used to absorb the impact of a fully loaded CCC from a height of 18 ft. The impact limiter consists of a 1.5-in. layer of aluminum honeycomb, with a compressive strength of 4,400 lb/in^2, and a CCC impact area of 153 in^2.

The dynamic finite-element analysis (WHC-SD-FF-DA-077) shows that the CCC will remain intact during and after the 18-ft vertical drop into the ISC, and the CCC boundary will provide adequate sealing during and after the drop. The results of the stress evaluation indicate that the stresses in some of the components exceed the yield strength but they are within the stress limits of the ASME Code, Section III, Appendix F (ASME 1989). The magnitude of the CCC tube plastic deformation does not affect the integrity of the CCC or the seals. The critical component of the CCC for membrane stress is the central tube of the CCC. The maximum membrane compressive elastic stress is 37,313 lb/in^2, which is less than the allowable 39,360 lb/in^2.

If an assembly is stored in the center tube of the CCC, it will rebound and hit the CCC cover and the bolts, and the seal region will remain elastic. The impact limiter over the central tube limits the load to the socket region of the CCC cover to 8,000 lbf. The metal O-ring seal of the CCC cover will have a maximum opening of 0.005 in., which is within the manufacturer's deformation limit to retain sealing effectiveness. The seal will neither lose contact with the cover nor with the upper support flange. The CCC cover bolts are subject to a tensile stress of 61,666 lb/in^2, which is less than the allowable 119,000 lb/in^2.

4-ft sideways drop of the CCC while inside the ISC. According to Document No. 910683, *FFTF Spent Fuel Interim Storage Cask Design Analysis Report* (General Atomics 1995), the CCC, inside the ISC, is subjected to a 96.1-*g* maximum side loading during a 4-ft drop. The ISC remains leaktight during this event. If it is assumed that the CCC rests on the ISC wall, the results show that the CCC material will not exceed the yield strength of the materials. Thus, no permanent deformation will occur.

If it is assumed that the CCC has 0.5 in. that can be deflected within the ISC liner, the dynamic finite-element analysis case shows that the elastic stresses in some of the components exceed the yield strength, but they are within the stress limits of the ASME Code, Appendix F, Section III, Division 1 (ASME 1989). The conditions of this analysis assume that the fuel element within the CCC hits the tube wall, and then the tube wall and element move together and hit the side of the ISC. The stress results show the bending stress in the storage tube is 51,608 lb/in^2, which is less than the membrane-plus-bending stress allowable of 59,040 lb/in^2. The magnitude of the plastic deformation is not expected to affect the integrity of the CCC because stresses are within allowable limits. Note that a 0.5-in. gap is not expected because the CCC will rest on the wall of the ISC during the drop.

4-ft ISC drop onto an unyielding surface. The 4-ft drop analysis consists of two separate drops: an end-drop and an angle drop with slap down. The maximum stresses on the ISC confinement barrier due to the 4-ft end-drop condition onto an unyielding surface occur in the 1.5-in. thick inner cylinder. A shell/ring discontinuity analysis was used to determine the stress at the upper flange-to-cylinder junction, and the maximum stresses were determined to be only 16.54 ksi. The *g*-level during a 4-ft drop onto an unyielding surface is between 70 *g* and 96 *g*. The stresses are within the normal condition allowable limits. Additionally, the bolt stress due to the flange rotation is within normal limits. Therefore, the ISC closure will not be affected during the accident conditions, and the sealing integrity is maintained.

The maximum calculated stress on the ISC confinement barrier occurs during the ISC slap down from an angled 4-ft drop. The maximum stress component is the inner 1.5-in. thick cylinder. The maximum stress that occurs for this condition is 17.4 ksi, and the maximum shear stress is 3.9 ksi. These stresses are within limits.

For the drop accidents, strength of material calculations following the methods of ACI-349, *Code Requirements for Nuclear Safety Related Concrete Structures*, indicate that the steel and concrete composite shielding sections of the cask will not be compromised during the drop events. The maximum amount of concrete crush during this accident is 6 in.

8-ft ISC drop onto a 1.5-ft thick concrete pad. The 4-ft drop analyses were conducted assuming a hard, unyielding surface. However, the ISA storage pad is a 1.5-ft thick concrete pad that will absorb much of the impact energy. To ascertain the effects of a yielding surface, an analysis was also performed for an 8-ft drop onto a 1.5-ft thick concrete pad. The highest *g*-level is 46 *g*. Analyses of these drops (SNF-4790) show that they are bounded by the 4-ft drop onto an unyielding surface.

Based on the drop scenarios postulated, confinement and geometry control are not lost for an 8-ft drop of the ISC package (including the CCC and fuel) onto the 200 Area ISA storage pad.

ISC drop onto another object. The ISC could be moved over other objects during handling. If the ISC drops onto an object, the ISC could strike the object and then tip over, resulting in a horizontal drop onto the 200 Area ISA storage pad. If the object struck by the ISC is taller than 4 ft, the horizontal drop will be outside of the bounds of the drop analyses. Therefore, the ISC is assumed to not be moved over objects that are taller than 4 ft. Note that the cask trailer is less than 4 ft tall.

40-in. ISC drop onto a 6-in. diameter mild steel punch. The ISC was evaluated for the hypothetical accident condition 40-in. drop of the ISC onto a 6-in. diameter mild steel punch. Conservatively, 1.3 times the weight for casks with diameters less than 30 in. was used. Only the concrete is used to stop the punch, even though there is enough steel to stop the punch also.

It was concluded that the ISC is of sufficient thickness to preclude punching shear failure. The primary stresses across the cask cross-section and the closure and bottom plates due to a puncture drop event are always less than 26.2 ksi, which are less than the allowable values of 63.3 ksi for primary membrane plus bending stress. The stainless steel closure, and bottom plate, and the cask body sidewall are of sufficient thickness to preclude punching sheer failure.

Neutron Radiography Facility TRIGA Fuel

The NRF TRIGA fuel could arrive at the ISA storage pad in either casks or DOT-6M containers, which are then placed into the Rad-Vault[4] storage container for long-term storage. During handling operations, an NRF TRIGA cask, DOT-6M container, or Rad-Vault could be

[4] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.

involved in a drop accident. Handling and drop analyses were performed to analyze the effects of drops. The following are the types of drops that could result in significant impact to one of the NRF TRIGA packages:

- Cask drop onto another cask or Rad-Vault
- Rad-Vault lid drop
- NRF TRIGA cask drop
- DOT-6M container drop.

Cask drop onto another cask or Rad-Vault. The cask drop onto another cask or Rad-Vault is bounded by the crane boom impact analysis discussed in Section D3.4.2.2, and is not addressed further in this section.

Rad-Vault lid drop. The Rad-Vault is not lifted while loaded with contents or with the lid in place according to *On-Site Storage Containers and RADVAULT Storage Containers Technical Information Package* (CNSI 1992). The Rad-Vault lid will be placed over the body after the cask or DOT-6M container is placed inside. The lid of the Rad-Vault, if dropped, will not drop onto the cask or DOT-6M containers stored in the Rad-Vault because the cylindrical design physically precludes this event (the outside diameter of the lid [114-in.] is much greater than the inside diameter [79-in.] of the vault body). Rad-Vault handling limits (TSR) preclude lifting the Rad-Vault lid more than 0.3 m (12 in.) above the top of the Rad-Vault in order to minimize potential structural damage to the Rad-Vault. Therefore, no scenario has been identified that would release radioactive material as a result of dropping the Rad-Vault lid.

Neutron Radiography Facility TRIGA cask drop. During handling operations, any one or all of the NRF TRIGA casks, DOT-6M containers, or the Rad-Vault could be involved in a drop accident that threatens the confinement. WHC-SD-FF-TI-043, *Safety Analysis Calculations for TRIGA Fuel Storage at 400 Area Interim Storage Area*, documents a handling and drop analysis that was performed to analyze the effects of a credible drop. In reality, the worst-case off-normal handling scenario would be the dropping of an NRF TRIGA cask onto one or more other casks. The Rad-Vault is in place before any fuel is moved, and the DOT-6M containers are the last to be loaded into the Rad-Vault and the first to be removed. A cask-on-cask drop is bounded by the crane boom drop accident. The lid of the Rad-Vault, if dropped, will not drop onto the fuel casks or containers already stored in the Rad-Vault because the cylindrical design physically precludes this event. The analysis demonstrates that under credible drop conditions, the NRF TRIGA casks and DOT-6M containers do not release radioactive material. The 10-ft/min set down load was equated to a 1-ft drop and determined to be bounded by the 109-in. NRF TRIGA cask drop without the impact limiters and the 30-ft DOT-6M container drop with impact limiters. However, to preclude exceeding the design criteria and the analyzed conditions, lifting limits have been imposed and are identified in Section D3.4.2.1.5.

DOT-6M container drop. Tests were performed on DOT-6M shipping packages (with 2R inner containers) to evaluate their response to a drop from 9 m (30 ft) onto an unyielding surface, and a 1-m (40-in.) drop onto a steel punch bar. A detailed observation of the packages following the drop testing showed the bolts to be tight and no damage to other areas that would indicate the possible loss of confinement. The results of the testing are reported in *Test Report of the 110 Gallon 6M-2R Shipping Package* (Martin Marietta 1992).

The two DOT-6M containers will have TRIGA fuel follower control rods (FFCRs) instead of the standard fuel payload. The only differences between the packages tested and the TRIGA FFCR configuration are the type of seal used in the 2R inner container, and the anticipated contents within the packages. The drop-tested packages used an elastomeric O-ring seal, whereas the TRIGA FFCR shipping packages will use a metallic O-ring seal to assure that long-term storage requirements are met. In addition, the drop-tested package used can-and-bottle internal containers, whereas the TRIGA FFCR shipping packages will contain stainless steel clad fuel rods enclosed in stainless steel pipe. The TRIGA FFCRs are not externally contaminated and the solid hydride fuel itself is enclosed within the stainless steel cladding, which has been demonstrated not to leak.

The loaded DOT-6M container has been shown to survive drops at heights greater than the NRF TRIGA cask. However, to ensure that administrative controls are easily promulgated into field operations, the administrative restriction for the NRF TRIGA cask maximum lift height is conservatively applied to the DOT-6M containers (21 in. above the top of the Rad-Vault, with the lid removed).

Light Water Reactor Fuel

A 250-ton crane is used to place the NAC-1 storage casks within the associated International Standards Organization (ISO) shipping containers at the 200 Area ISA. The worst-case handling or drop scenario is for the NAC-1 cask to be dropped during cask handling or transfer operations. Administratively, only one cask/container will be handled at a time, and maximum lift heights are imposed based on this analysis.

The NAC-1 casks are transported and placed at the 200 Area ISA while inside ISO containers. The presence of the NAC-1 cask within the ISO shipping container was not considered in the NAC-1 Safety Analysis Report (SAR) bounding cask drop scenario. However, the cask tie-down devices (cask rotation trunnions) are designed to fail under excessive loads in the trunnion tubes, such that cask integrity and shielding are not affected. During the postulated drop events, the cask mounted within the ISO shipping container provides additional energy absorption capabilities that were conservatively ignored in the accident analysis. Thus, the NAC-1 cask drop is assumed to be bounding over the ISO container drop.

The types of handling and drop impacts analyzed for the NAC-1 cask are as follows:

- Drop onto unyielding surface in various orientations
- Drop onto steel rod
- Drop onto another object.

Drop onto unyielding surface in various orientations. Lawrence Livermore National Laboratory document UCID-21246, *Dynamic Impact Effects on Spent Fuel Assemblies*, assessed the effects of dynamic impacts (to be expected from cask drop or tipover incidents) on the integrity of the fuel rod cladding for zircaloy-clad LWR spent fuel assemblies during cask handling and storage. The conclusions of this report define acceptable *g*-load limits for fuel assemblies internal to the package and state:

> "The analysis of the capability of spent fuel rods to resist impact loads caused by storage cask accidents indicates that, for the most vulnerable fuel assembly, axial

buckling varies from 82 *g* at initial storage to 95 *g* after 20-year storage. In a side drop, no yielding is expected below 63 *g* at initial storage to 74 *g* after 20-year storage. For storage casks designed to limit loads at or below these *g* levels, it is not likely that damage will occur to the spent fuel rods."

Individual fuel rods stored in the NAC-1 casks are packaged in a rod consolidation container that retains the cross-sectional geometry within the inner canisters. The individual rods therefore remain oriented similar to the rods in the fuel assemblies and are subject to similar buckling loads and conclusions.

It is permissible to store damaged LWR fuel (does not meet "intact" fuel inspection criteria, see Section 2.5.1) in the double welded LWR canister because the leaktight container prevents water intrusion, which precludes the possibility of criticality even without geometry control within the canister.

The drop onto an unyielding surface also considers potential impacts to the shielding associated with the NAC-1 cask. The drop analyses (NAC-E-804) consist of a postulated 30-ft drop onto an unyielding surface in the following orientations:

- Top end impact
- Top corner impact
- Side impact
- Bottom end impact
- Bottom corner impact.

Top end impact. The top end impact analysis focuses on the crush strength and crush area of the balsa wood in the upper impact limiter, and assumes that the balsa crush strength and crush area are constant during the impact event. The analysis also investigates the bending of the cask ring structure to ensure that the ring structure has sufficient rigidity as a base for the compressive loads of the balsa. This analysis concludes that the stress of the material in the ring structure is well below the yield strength and will provide a sufficient base (without yielding) for the impact limiter to crush against. The analysis also determined the impact deceleration loads for the top impact condition would not exceed 44.67 *g*, which is within the 82 *g* allowable.

Bottom end impact. The analysis for the bottom end impact assumes that the impact energy transferred to the bottom of the cask is attenuated by the lower ring structure. During normal operation, the lower ring structure is designed with enough rigidity to be used as a pedestal when the cask is in the upright configuration. However, it also functions as a sacrificial structure that deforms to dissipate impact energies. The representative analysis of the impact indicates that the bounding deceleration load is 76.6 *g*, which is within the allowable load of 82 *g*.

This calculated information was used as an input value to determine if the lead shielding (located between the inner and outer cask shell) settles or slumps, based on the magnitude of the deceleration load. The analysis determined that the 1-in. step provided in the top flange between the shells is adequate to keep the lead from slumping. This analysis neglected the resistance imposed by the copper heat transfer fins and the radial support provided by the inner and outer stainless steel shells. It was also determined that the lead will not yield in direct shear to allow the lead to slump by shear into a void region left by the original placement melt.

Side impact. Impact on the side of the cask is attenuated by deformation of the top and bottom ring structures, the expansion chamber, and the neutron shield chamber. Since the neutron shield chamber is empty, these four components are considered to be sacrificial. The analysis documented the energy capacity of the various components separately and plotted the values versus a common parameter. The composite energy absorption capacity of the structure was then equated to the total impact energy to arrive at the deformation impact to the sacrificial structures. From this approach, it was determined that the bottom ring structure will deform approximately 8.75 in. and the top ring structure will deform 7.5 in., absorbing the impact energy as designed, which results in an adequate and acceptable condition.

The ring structure was analytically checked for buckling in the side drop analysis. The analysis determined that the critical buckling stress of the inner portion of the ring structure is in excess of the dynamic compressive yield strength of the material. Therefore, compression yielding precludes buckling, and the side impact deformation analysis is adequate.

The side impact deceleration load was determined in order to evaluate whether the load imposed by the lead shielding on the inner stainless steel liner is great enough to cripple the liner. The deceleration load was determined to be 96 *g* on the lead between the inner and outer shells. The analysis assumed that the lead would act as a fluid. As such, it was determined that the hydrostatic pressure on the inner shell imposed by the lead will be 260 lb/in^2. Analyses in the NAC-1 SAR for the contraction of molten lead around the inner shell during cask fabrication indicate that the inner shell withstands hydrostatic pressures of up to 506 lb/in^2. Thus, the 260 lb/in^2 imposed by the lead during the drop is within allowable pressures for the inner shell.

The side impact deceleration load was also compared against the allowable *g*-loads for the fuel assemblies. The 96 *g* deceleration load includes a peaking factor of 2.0, multiplied in the equation to calculate the force on the lead shielding and the inner shell due to the lead. The acceleration within the inner shell does not require the peaking factor since it was used to consider local effects at the end stiffening rings, which are subject to large deformation. These deformations were calculated to be 8.75-in at the lower ring and 7.5-in at the top ring. The maximum *g* load is calculated as the drop height (30 ft or 360 in.) divided by the minimum crush distance (7.5 in). Therefore, $G = 360/7.5 = 48\ g$ applied to the fuel assemblies for the side drop of the NAC-1 cask. This is below the 63 *g* load required to cause yielding of the fuel assembly; therefore, no loss of structural integrity is expected for the side drop of the NAC-1 cask.

Top corner impact. This analysis investigates the loads imposed on the top of the cask structure when subjected to a top corner impact. Component calculations include the shearing resistance of the top impact limiter, the top corner impact deceleration loads, and the top corner impact bolt stress. The calculation determines that the loads imposed by the top corner impact are less than the bolt preload stress, which results in an adequate and acceptable condition.

Bottom corner impact. The bottom corner impact energy was analyzed assuming the maximum damage to the cask would be in the area where a drain valve is directly over the point of impact. The kinetic energy in the cask was assumed to primarily be dissipated by the major impact limiter components (i.e., the balsa, the shock tube, the gusset plates and the fin plates). Load models were developed to determine deformation loads and dissipation energies. The analysis concluded that cask kinetic energy can be absorbed by deformation of the sacrificial components external to the base of the cask that contains and surrounds the drain penetrations and minor deformation of the cask boundary. This area was considered safe during this design impact loading.

For the 30-ft drop in various orientations onto an unyielding surface, the NAC-1 SAR analyses conclude that the confinement and shielding integrity of the NAC-1 cask will be maintained, although significant damage will occur to the exterior, sacrificial structures of the cask.

Drop onto steel rod. This accident scenario evaluated the cask free-fall of 40-in. onto the end of a mild steel pin to ensure that no loss of confinement occurs. The analysis (NAC-E-804) evaluated three locations determined to be the most likely for maximum damage to occur. These locations are as follows:

- Impact on the midspan of the cask body
- Impact of the side of the cask lid
- Direct impact on a valve.

For the impact on the midspan of the cask body, the outer shell of the cask was analyzed in relation to the 40-in. free drop. Dropping the cask such that the pin impacts the midsection of the outer shell imposes the greatest bending moment into the cask. The analysis determined that the maximum stress is small compared to the allowable stresses and that little, if any, damage to the cask will result from this drop.

The second analysis investigated the impact of the pin on the cask lid from the 40-in. drop height. The analysis determined that the cask lid bolts are not loaded in shear since the clearance between the cask lid and flange is less than the bolt hole clearance. The analysis concluded that the cask lid would be radially displaced 4.4×10^{-3} in. at the flange seal interface. Since the NAC-1 cask is not a confinement boundary, this result has no safety consequence.

The third analysis for this section evaluated the pin loading from the 40-in. free drop of the cask on a confinement valve. The valves are recessed within the stainless steel flanges at the closure (top) end of the cask. The valves do not protrude from the cask nor are they located in trunnions. This design approach provides protection for the valves by the inherent stability of the flange sections in this hypothetical accident scenario. The analysis of the valve impact on the steel rod concluded that steel flanges will adequately protect the valves.

For the 40-in drop onto a steel rod, the NAC-1 SAR analyses concluded that the confinement and shielding integrity of the NAC-1 cask will be maintained. Only minimal damage to the cask, namely the localized deformation and slight puncture of the shield tank, will be incurred.

Drop onto another object. The analysis for the drop of the NAC-1 cask does not consider impact onto another object and subsequent potential slapdown. Therefore, it is assumed that the NAC-1 casks will not be lifted over other objects except the transport trailer.

D3.4.2.1.2 Source Term Development.

For handling and drop accidents, there is no source term from any of the fuel packages for the events postulated and analyzed in the scenario development for any of the package types. However, the events and analysis depend on initial conditions that could be violated during operations. For example, the ISC has been shown to survive an 8-ft drop onto the 200 Area ISA pad, but a drop from a higher distance will represent an unanalyzed condition. If a drop occurs from a distance higher than those analyzed, catastrophic damage is not expected because of the margin and conservatisms in the analyses. But since the package responses are unknown, a drop from heights greater than those postulated in the scenario development are bounded by the source terms calculated for the worst-case, non-mechanistic release shown in Section D3.4.1 for each fuel type.

D3.4.2.1.3 Consequence Analysis.

All of the package types have been shown to maintain confinement of the fuel for the handling and drop events postulated and analyzed in the scenario development. In addition, the analyses have shown that shielding is maintained. For the events analyzed, there are no onsite or offsite releases of hazardous materials and no direct radiation hazards to workers greater than those expected during normal operations. A consequence analysis is not needed for the FFTF, TRIGA, or LWR fuel storage systems.

D3.4.2.1.4 Comparison to Guidelines.

Unmitigated event – For drops that exceed the initial conditions of the analyses, the bounding consequences for each fuel type show that the offsite release limits are not challenged. Therefore, no safety-class confinement SSCs are necessary.

For the NRF TRIGA fuel, the unmitigated onsite consequences do not exceed the onsite risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following a handling or drop accident, no safety-significant SSCs are required to confine the NRF TRIGA fuel.

For the FFTF fuel and the LWR fuel in the NAC-1 casks, the bounding onsite consequences for drops outside of the initial conditions analyzed exceed the onsite risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

While no safety-class SSCs are required to provide confinement, some SSCs are required to protect the geometric configuration of the fuel for criticality safety. SSCs credited with retaining the critically safe geometry of the fuel are designated as safety class.

Mitigated event – For the handling and drop events postulated and analyzed in the scenario development, the potential releases are prevented. These events take credit for the packaging components and limitations on handling (e.g., lift height) that have been shown to provide the impact protection and confinement of the fuel for the analyzed drop events.

D3.4.2.1.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls.

The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from inadvertent nuclear criticality during and following a drop event.

Fast Flux Test Facility Fuel

The following SSCs are designated as safety significant or safety class:

- **FFTF ISC** – The ISC is designated as safety significant to maintain confinement of radioactive materials after a credible drop and to provide passive protection of the CCC such that it retains structural integrity.
- **CCC** – The CCC is designated as safety class to maintain structural integrity to provide criticality geometry control.

The following TSR controls are necessary to protect the assumptions made in the drop analyses:

- **Lift height restriction** – The ISC cannot be lifted more than 8 ft from the surface of the 200 Area ISA pad.
- **Lift over object restriction** – The ISC cannot be lifted over an object that is taller than 4 ft or contains radioactive material.

Neutron Radiography Facility TRIGA

No safety-class or safety-significant SSCs are required to confine the potential release from the NRF TRIGA fuel based on the unmitigated analysis provided in Section D3.4.1. The NRF TRIGA casks and the DOT-6M containers do provide physical protection of the SNF in the event of a drop, and are therefore designated as important-to-safety Category B for NRC equivalency. These general-service containers were originally designed and procured in an NRC-equivalent manner appropriate for their planned contents. These items are currently in use providing storage of TRIGA fuel at the 400 Area ISA. The risk of continued use of the general-service TRIGA fuel containers at the 200 Area ISA is considered acceptable due to the low activity fuel, as discussed in the Section D3.4.1 analysis section. Therefore, there are no safety-class or safety-significant SSCs related to NRF TRIGA fuel storage.

The following TSR controls are necessary to protect the assumptions made in the drop analyses:

- **Lift restriction** – Based on the assumptions made in the drop analyses, the NRF TRIGA casks and DOT-6M containers shall not be lifted more than 109 in. (approximately 9.1 ft) from the surface of the ground at the 200 Area ISA.
- **Lift restriction** – The Rad-Vault lid shall not be lifted more than 12 in. above the top of the Rad-Vault.

Light Water Reactor Fuel

The following SSCs are designated as safety significant or safety class:

- **NAC-1 cask** –The NAC-1 cask is designated as safety significant to provide passive protection of the inner canister such that it retains structural integrity after a credible drop.

- **LWR canister** – The LWR canister is designated as safety significant to maintain confinement of radioactive materials after a credible drop and safety class to maintain structural integrity for criticality geometry control and a leaktight configuration to preclude water intrusion.

The following TSR controls are necessary to protect the assumptions made in the drop analyses:

- **Lift restrictions** – The NAC-1 package cannot be lifted more than 30-ft from the surface of the 200 Area ISA storage pad.
- **Lift over object restriction** – The NAC-1 package cannot be lifted over other objects except the transport trailer.

D3.4.2.2 Mobile Crane Fall.

D3.4.2.2.1 Scenario Development.

While handling the storage casks, an accident could possibly occur. The casks will be picked up and transferred from current locations at the 400 Area ISA and the 300 Area, and set down at the 200 Area ISA by a substantial crane. It is likely that different mobile cranes will be used in the different areas. The FFTF crane is a Manitowoc 4000W crawler, and the 200 Area ISA crane will be a Manitowoc 250T mobile crane. The 250T Manitowoc crane is considered bounding, and analyses were performed based on the structural components of this crane.

A mechanical failure could initiate a drop of a crane boom onto the SNF storage systems. If the impact caused forces beyond the design strength, the cask containment could be breached. Breach of the cask containment could result in radiological releases. The casks also protect the inner canister, which maintains the geometry of stored SNF to meet any criticality safety concerns. The crane boom was demonstrated to buckle under the lateral impact load before it could impart sufficient energy to breach the cask systems. The crane boom analysis is documented in SNF-4794, *200 Area Interim Storage Area Design Basis Accident Analysis Documentation for LWR Fuel*. A summary of the safety features required to prevent the mobile crane fall accident is provided in Table D3-12.

Fast Flux Test Facility

Previous 400 Area ISA analyses (WHC-SD-FF-DA-078, *Structural Analysis of Interim Storage Area [ISA] Concrete Pad and Crane Drop on Interim Storage Cask [ISC]*) were performed to assess the damage that a crane boom drop onto the ISC could have on its containment function. The analysis assumed that the crane was a Manitowoc 4000W crawler and stated that the results required additional evaluation for any other crane or for the same crane at a different elevation above the top of an ISC. A supplemental analysis has been prepared for the 200 Area ISA that evaluates the effects of using the 250T Manitowoc crane at the 200 Area ISA (SNF-4794).

Conservative analysis shows significant damage to the outer concrete of the ISC from the crane boom drop or crane hook drop, but the inner steel confinement and shields will not be damaged by the event. Under these conditions, the ISC will not credibly lose its confinement.

Table D3-12. Summary of Safety Features Required to Prevent Mobile Crane Fall Accident.

Candidate accident	Checklist designator[a]	Safety function	Safety features (described in Chapter D4.0)
Drop of a boom onto the cask	G-03	Maintain confinement of radioactive materials after the crane fall.	Safety-significant equipment for confinement: • FFTF ISC. Safety-significant equipment for structural integrity: • NAC-1[b] cask system • FFTF ISC. TSR: • Restriction on use of crane and lifting equipment. Defense-in-depth: • Qualified crane operators • Detailed procedures.

[a] Checklist designators are from SNF-4820, 1999, *200 Area Interim Storage Area Final Hazard Analysis Report*, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.
[b] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

FFTF = Fast Flux Test Facility.
ISC = interim storage cask.
TSR = Technical Safety Requirement.

Neutron Radiography Facility TRIGA

A mobile crane fall event was assessed that assumed a mechanical failure of the crane that resulted in the boom falling onto the Rad-Vault. The Rad-Vault and NRF TRIGA cask are general-service items, so no credit was taken for the protection they provide to the fuel assemblies. The Rad-Vault is 2.9 m (114 in.) in diameter by 2.8 m (111 in.) high with 2-ft thick concrete walls and a 2-ft thick concrete lid. Although structural failure of the Rad-Vault would not be anticipated due to its robust construction, no structural analysis was performed and therefore, cask damage was assumed. Failure of the crane would only impact a fraction of the fuel assemblies. It is not expected that the full length of any assembly would be crushed. It is assumed that the isotope activity of the entire inventory of the Rad-Vault (101 fuel elements) is equal to 101 times the isotope activity of the maximum irradiated fuel element. The assumptions used to model the unmitigated release bound this potential boom drop accident scenario and are within guidelines. No features are credited to prevent or mitigate this TRIGA fuel accident.

Light Water Reactor Fuel

An analysis was performed to determine that the impact load on the NAC-1 storage cask from the postulated crane boom drop was insufficient to breach the cask confinement boundary structure (SNF-4794). The LWR canister and internal components are therefore protected from damage. The analysis compared the boom drop accident to the cask 30-ft drop and 40-in. puncture drop accident events that were previously analyzed in the NAC-1 SAR (NAC-E-804). The presence of the ISO shipping container was not included in the analysis. This assumption is conservative since the shipping container is assumed to act as protection for the cask by absorbing some of the energy and distributing the load imparted to the cask during the crane boom drop. The analysis of the crane boom drop concludes that the crane boom will fail before it can impart sufficient load on the cask to breach confinement. Local lacing members in the boom fail at the point of impact and the boom structure collapses, relieving the load and subsequent energy imparted to the cask.

Cask Energy Absorption. The response of the cask to the 30-ft side impact drop is appropriate for comparison to the boom drop effects. In a side drop, the SAR analyses show that the cask top and bottom end ring support structures, along with the neutron shield chamber, can absorb approximately 23×10^6 in-lb of energy without breach of confinement. These members are considered totally sacrificial structures and energy absorption is achieved by means of considerable crushing of these structures. The crane boom is 80 ft long and weighs 22,490 lb. The potential energy of the boom falling from the vertical position is 11×10^6 in-lb. The reaction forces on the cask at the end ring support structures for the crane boom drop are therefore enveloped by the existing 30-ft drop accident analysis.

Cask Bending Moment. The maximum bending moment of the cask was also evaluated during the crane boom drop event and compared with the bending moment imparted to the cask during the 40-in. puncture drop (SNF-4794). Impact on the midspan of the cask was considered the most critical point for beam-bending of the cask. The analysis of the cask bending moment during the crane boom drop shows that the maximum reaction force the boom can impart on the cask during impact is 2.9×10^5 lb at the mid-point of the crane boom. This value compares with an allowable force of 6.8×10^5 lb that would be required to impart the equivalent bending moment analyzed in the 40-in. drop analysis. Thus, the bending moment of the cask during the crane boom drop event is approximately a factor of two times less than that found acceptable in the previous NAC-1 SAR analysis.

Cask Puncture Resistance - Crane Boom. The cask's resistance to puncture during the crane boom drop event was also analyzed (SNF-4794). The analysis concluded that the maximum reaction force that the crane boom could impart to the cask was 2.9×10^5 lb. This compares to the calculated value of 6.9×10^5 lb required for the bottom cord members of the crane boom to puncture the cask's 1¼-in. outer shell. Thus, puncture of the cask's outer shell by the crane boom is not expected.

Cask Puncture Resistance - Load Block. An analysis was performed to evaluate the ability of the 4,000-lb load block at the end of the crane boom to puncture the cask (SNF-4794). The load block has a massive hook with about a 2.5-in. leading edge that results in an initial contact area between the block and the cask that is less than the area of the 6-in. diameter pin in the 40-in. drop accident. The maximum lift height of the load block above the cask is 75 ft. The results of the analysis conclude that although the cask will suffer physical damage, the integrity of the cask will be maintained in the maximum block height condition. Conservatively, a value of 56 ft will be used as limiting criteria for the block height over a cask/ISO container system (SNF-4794). This height provides sufficient overhead clearance to perform all rigging and hoisting activities at the ISA.

Cask Puncture Resistance - Hydraulic Spreader Bar. An analysis was performed to evaluate the capability of the 20,610-lb hydraulic spreader bar, connected at the crane hook, to puncture the cask if accidentally dropped (SNF-4794). The hydraulic spreader bar is a massive lifting fixture for the ISO containers. It has a frame of large tubular channels that allow it to telescope in length. It also has attached positioning guides at each corner that protrude downward 15 in. beneath the corner connecting devices. These alignment plates are 0.75 in. thick by 8 in. across. The analysis evaluated bending loads from a drop accident where the fixture lands at a single point on its side, and then assessed the puncture potential from a flat drop that lands on a single alignment plate as the worst-case leading edge for the puncture. The results of the analysis conclude that bending is the controlling parameter that establishes the requirement for a maximum permissible lift height. Although the cask may suffer physical damage, the confinement boundary of the cask will be maintained (protecting the LWR canister and internal components from damage) by imposing a maximum height condition of 6 ft above the ISO container lid, approximately 14.75 ft above the ground (SNF-4794). This height provides sufficient overhead clearance to perform all rigging and hoisting activities at the ISA.

The analysis of crane boom, load block drop, and hydraulic spreader bar drop scenarios do not take credit for the presence of the ISO shipping container. The cask, as mounted in the ISO shipping container, is designed to shear at the lifting and rotation trunnions during an impact event, while maintaining cask integrity (NAC-E-804). Since the ISO shipping container is not required to survive the crane boom and load block drop accidents, it provides no safety function for this accident.

D3.4.2.2.2 Source Term Analysis.

Current analyses for the FFTF and LWR NAC-1 casks demonstrate that complete confinement of the SNF will be maintained during and after the analyzed crane boom fall. No source term analysis is required for the ISC or NAC-1 cask.

The source term analysis developed for the unmitigated TRIGA accident in Section D3.4.1.1 is applicable for this scenario.

The events and analysis depend upon initial conditions; however, these conditions can be violated during operations. For example, a crane might be taken out of service and replaced with a different crane, which would represent an unanalyzed condition. If a boom drop were to occur with a different boom, catastrophic damage is not expected because of the margin and conservatisms in the analyses. But since the package responses are unknown, a boom drop from a crane that has not been analyzed in the scenario development is considered bounded by the source terms calculated for the worst-case, non-mechanistic release shown in Section D3.4.1 for each fuel type.

D3.4.2.2.3 Consequence Analysis.

The postulated crane fall accident consequences have been prevented by the passive design features of the ISC and the NAC-1 cask. A consequence analysis is not needed for the FFTF or LWR NAC-1 casks.

The consequence developed for the unmitigated TRIGA accident in Table D3-10 is applicable for this scenario.

D3.4.2.2.4 Comparison to Guidelines.

Unmitigated event – For events that exceed the initial conditions of the analyses, the bounding consequences for each fuel type show that the offsite release limits are not challenged. Therefore, no safety-class confinement SSCs are necessary.

For the NRF TRIGA fuel, the unmitigated onsite consequences do not exceed the onsite risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following a handling or drop accident, no safety-significant SSCs are required to confine the NRF TRIGA fuel.

For the FFTF fuel and the LWR fuel in the NAC-1 casks, the bounding onsite consequences for drops outside of the initial conditions analyzed exceed the onsite risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

Mitigated event – The postulated crane fall accident consequences have been prevented by the passive design features of the FFTF and NAC-1 casks; therefore, there is no release from this event.

For TRIGA SNF, the consequence analysis was performed for the 200 Area ISA based on the unmitigated release discussed in Section D3.4.1.2. Using appropriate air transport and unit dose factors, this particulate has been used to calculate onsite and offsite receptor dose estimates. The total offsite and onsite individual doses are estimated to be 3.4×10^{-6} rem and 9.01×10^{-3} rem, respectively.

The radiological dose consequences from the crane fall accident for the 200 Area ISA are below the offsite release limits and the onsite dose consequence guidelines; therefore, no safety-class or safety-significant equipment is required to mitigate the potential effects of this accident for TRIGA fuel.

D3.4.2.2.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls.

The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from mobile crane fall accidents.

Fast Flux Test Facility Fuel

The following SSC is designated as safety significant:

- **FFTF ISC** – The ISC is designated as safety significant to maintain confinement of radioactive materials after the crane fall and to provide passive protection of the CCC such that it retains structural integrity.

The following TSR control is necessary to protect the assumptions made in the mobile crane fall analyses:

- **Lift restriction** – A TSR shall be in place to restrict crane use to that bounded by the calculational analysis for the FFTF ISC. Only the Manitowoc 4000 150T crane with Model 22 80-ft boom or the Manitowoc 4100 250T crane with Model 27 80-ft boom shall be used to lift the ISC (SNF-4794).

Neutron Radiography Facility TRIGA

No SSCs credited with a preventive or mitigative function.

Light Water Reactor Fuel

The following SSC is designated as safety significant:

- **NAC-1 cask system** – The NAC-1 cask has been designated safety significant to provide passive structural integrity protection of the inner canister.

The following TSR control is necessary to protect the assumptions made in the mobile crane accident:

- **Lift restriction** – A TSR shall be in place to restrict use of the crane, lifting equipment, and lift height to that bounded by the calculational analysis for the NAC-1 casks. Only the Manitowoc 4000 150T crane with Model 22 80-ft boom or the Manitowoc 4100 250T crane with Model 27 80-ft boom shall be used to lift the ISC (SNF-4794).

D3.4.2.3 Cask Tipover.

D3.4.2.3.1 Scenario Development.

Cask handling, operations near the storage casks, or external forces could cause an accident that would lead to the tipover and slap-down of a cask. If the impact caused a cask to undergo accelerations and forces beyond the design strength, the cask containment could be breached. Breach of the cask containment could result in radiological releases. Casks also maintain the geometry of stored SNF to meet any criticality safety concerns, and a cask breach or deformation could lead to the violation of a criticality control limit. A summary of the safety features required to prevent or mitigate the cask tipover accident is provided in Table D3-13.

Table D3-13. Summary of Safety Features Required to Prevent Cask Tipover Accident.

Candidate accident	Checklist designator[a]	Safety function	Safety features (described in Chapter D4.0)
Cask tipover	G-03 G-06 G-13	Maintain criticality geometry control after cask is overturned. Maintain confinement of radioactive materials and provide passive protection such that structural integrity is maintained after a cask is overturned.	Safety class equipment for criticality geometry control: • CCC. Safety-significant equipment for confinement: • FFTF ISC. Safety-significant equipment for structural integrity: • FFTF ISC • ISO container • NAC-1[b] cask. Defense-in-depth: • Rad-Vault[c] • NRF TRIGA[d] cask • DOT-6M[e] container • 2R container[f] • Qualified crane operators • Detailed procedures.

[a] Checklist designators are from SNF-4820, 1999, *200 Area Interim Storage Area Final Hazard Analysis Report*, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.
[b] NAC-1 casks are manufactured by Nuclear Assurance Corporation.
[c] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.
[d] TRIGA is a trademark of General Dynamics Corporation.
[e] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.
[f] 2R designates the inner container of DOT-6M.

CCC = core component container.
FFTF = Fast Flux Test Facility.
ISC = interim storage cask.
ISO = International Standards Organization.
NRF = Neutron Radiography Facility.

Fast Flux Test Facility

The ISC design analysis report (General Atomics 1995) describes the cask tipover accident analyses performed for the ISC. The analysis considers an unrestrained rotational fall from a vertical to a horizontal position onto a flat unyielding surface. This analysis demonstrates that the ISC will survive a tipover accident onto any flat horizontal surface. The cask forces during this accident are bounded by those for the accident analysis for a 4-ft free drop of the cask side onto an unyielding surface (General Atomics 1995). Under these conditions, the ISC will not credibly lose its confinement and criticality control geometry and, thus, will not lead to credible radioactive releases.

Neutron Radiography Facility TRIGA

The 200 Area ISA analysis performed for the Rad-Vault containing six NRF TRIGA casks and two TRIGA FFCR DOT-6M shipping containers is documented in SNF-4792, *200 Area Interim Storage Area Design Basis Accident Analysis Documentation for TRIGA® Fuel*. The analysis demonstrated that the Rad-Vault will not tip over during a 360 mph design basis tornado (DBT) or seismic event.

While no analysis specifically addresses a TRIGA cask tip over, drop analyses have been performed that would bound such an accident. The TRIGA cask has been analyzed for a 109-in. drop height onto concrete, with the cask in a horizontal position. The concrete used in the analysis has a thickness of 16 in., a strength of 4000 lb/in^2, and is reinforced with #7 rebar that is spaced 14 in. apart and placed 2 in. below the concrete surface. A simple unrestrained tipover accident of a TRIGA cask onto any flat surface, with compressive strength less than or equal to the described concrete, is not expected to lead to a breach of cask confinement.

While no tipover testing was performed for the DOT-6M, severe drop tests were completed that will bound such an accident. The drop tests onto an unyielding surface showed no breach of confinement for a 30-ft drop. These tests were conducted with the container in a horizontal orientation (Martin Marietta 1992). Any tipover accident onto a flat horizontal surface will be bounded by the results of this horizontal drop test, which demonstrated that containment will not be breached. Under these conditions, the DOT-6M will not credibly lose its confinement and will not lead to credible radioactive releases.

Light Water Reactor Fuel

While an unrestrained tipover accident has not been analyzed, the NAC-1 cask was analyzed previously for a 30-ft free drop onto a flat, horizontal unyielding surface for end, side, and corner impact orientations (NAC-E-804, Section 2.7.1). The NAC-1 cask will not breach from this 30-ft drop for any orientation, and the analysis is expected to easily bound any consequences of a simple unrestrained tipover onto any flat surface. Under these conditions, the NAC-1 cask will not credibly allow damage to the LWR canister or internal components resulting in loss of criticality control geometry, loss of leaktight configuration, or credible radioactive releases.

D3.4.2.3.2 Source Term Analysis.

Fast Flux Test Facility

All ISA analyses demonstrate that complete confinement of the SNF will be maintained during and after the postulated and analyzed tipover events. All current analyses assumed that the impact surface is unyielding. The ISA storage pad will yield to the impact of an FFTF cask drop, and absorb drop energy at least as well as the analyzed concrete. The tipover or drop analysis assumes, in general, that the drop surface is flat. The ISA storage pad and any areas that a cask could tip over onto should be flat. The confinement of each cask will be maintained in each of the considered tipover events and, therefore, a source term analysis is not required.

Neutron Radiography Facility TRIGA

The Rad-Vault has not been analyzed for a tipover accident, but has been demonstrated to not tip during a DBT or seismic event. A tipover accident is not considered credible for the Rad-Vault in the 200 Area ISA.

Light Water Reactor Fuel

The existing ISA analyses demonstrate that complete confinement of the SNF will be maintained during and after the postulated and analyzed tipover events.

D3.4.2.3.3 Consequence Analysis.

Fast Flux Test Facility

The postulated tipover accident consequences have been prevented by the passive design features of the FFTF ISC. A consequence analysis is not needed.

Neutron Radiography Facility TRIGA

The postulated tipover accident consequences have been prevented by the passive design features of the Rad-Vault and each TRIGA container. A consequence analysis is not needed.

Light Water Reactor Fuel

The postulated tipover accident consequences have been prevented by the passive design features of each NAC-1 cask. A consequence analysis is not needed.

D3.4.2.3.4 Comparison to Guidelines.

Unmitigated event – For scenarios with conditions that exceed the initial conditions of the analyses, the bounding consequences for each fuel type show that the offsite release limits are not challenged. Drop analyses have been performed that would bound the cask tipover accident. Therefore, no safety-class confinement SSCs are necessary.

For the FFTF fuel and the LWR fuel in the NAC-1 casks, the bounding onsite consequences for drops outside of the initial conditions analyzed exceed the onsite risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

For the NRF TRIGA fuel, the unmitigated onsite consequences do not exceed the onsite risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following a tipover accident, no safety-significant SSCs are required to confine the NRF TRIGA fuel.

While no safety-class SSCs are required to provide confinement, some SSCs are required to protect the geometric configuration of the fuel for criticality safety. SSCs credited with retaining the critically safe geometry of the fuel or a leaktight configuration are designated as safety class.

Mitigated event – For the tipover events postulated and analyzed in the scenario development, the potential releases are prevented. These events take credit for the packaging components and limitations on handling (e.g., lift height) that have been shown to provide the impact protection and confinement of the fuel for the tipover drop events.

D3.4.2.3.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls.

The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from inadvertent nuclear criticality during and following a tipover event.

Fast Flux Test Facility Fuel

The following SSCs are designated as safety significant or safety class:

- **FFTF ISC** – The ISC is designated as safety significant to maintain confinement of radioactive materials after the tipover and to provide passive protection of the CCC such that it retains structural integrity.
- **CCC** – The CCC is designated as safety class to maintain structural integrity to provide criticality geometry control.

Neutron Radiography Facility TRIGA

The Rad-Vault has been analyzed to not tip over or slide as a result of natural phenomena hazards or DBAs, and the NRF TRIGA casks and DOT-6M containers are not expected to leak as the result of a tipover event. No safety-class or safety-significant SSCs are required to confine the potential release from the NRF TRIGA fuel based on the unmitigated analysis provided in Section D3.4.1.

Light Water Reactor Fuel

The NAC-1 cask in the ISO container has been analyzed to not tip over or slide as a result of natural phenomena hazards or DBAs.

D3.4.2.4 Fuel Rod Rupture.

D3.4.2.4.1 Scenario Development.

Fission gases generated in the SNF are contained by the fuel cladding. Deterioration of the clad by oxidation or stress cracking could result in a release of the contained fission gas to the storage container. If the design pressure value for the sealed storage container is exceeded, these fission gases could be released to the environment. A summary of the safety features required to prevent the fuel-rod rupture accident is provided in Table D3-14.

Table D3-14. Summary of Safety Features Required to Prevent Fuel Rod Rupture Accident.

Candidate accident	Checklist designator[a]	Safety function	Safety features (described in Chapter D4.0)
Uncontrolled release of gases from fuel rod rupture from casks	H-06 H-11	Maintain structural integrity for criticality geometry control and leaktight configuration after pressurization. Maintain confinement after rupture of all fuel pins.	Safety class equipment for criticality geometry control and leaktight configuration: • CCC • LWR canister. Safety-significant equipment for confinement: • FFTF ISC • LWR canister. Defense-in-Depth: • NRF TRIGA[b] cask • 2R container.[c] TSRs: • Restriction on minimum spacing between ISCs • Restriction on maximum fuel loading in the FFTF ISC and LWR canister.

[a] Checklist designators are from SNF-4820, 1999, *200 Area Interim Storage Area Final Hazard Analysis Report*, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.
[b] TRIGA is a trademark of General Dynamics Corporation.
[c] 2R designates the inner container of DOT-6M. (DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation)

CCC = core component container.
FFTF = Fast Flux Test Facility.
ISC = interim storage cask.
LWR = light water reactor.
NRF = Neutron Radiography Facility.
TSR = Technical Safety Requirement.

Fast Flux Test Facility Fuel

Based on the ISC design analysis report (General Atomics 1995, page 4.1-4), the cavity internal pressure for normal long-term conditions is 33.9 lb/in^2 gauge, based on an average fuel cavity temperature of 245 °F. According to WHC-SD-FF-ER-100, *Thermal Analysis of the Core Component Container Within the Interim Storage Cask*, the maximum pressure occurs for volcanic ash accumulation with plugged ducts and is 36.7 lb/in^2 gauge based on the highest fuel cavity temperature of 286 °F. These pressures are calculated at the end of the cask's 50-year life assuming fission gas generation and no reduction in decay heat. Per Specification WHC-S-4110, *Specification for FFTF Interim Storage Cask*, Section 3.2.2.3, the generation of gases produced by radioactive decay is expected to be 5.4 gram-atoms (moles) per assembly or 37.8 gram-atoms per ISC for a 50-year period, assuming that all free gases contained within the fuel rod cladding are released. The design pressure is based on a CCC containing seven fuel assemblies.

These results have assumed that the casks are spaced no closer than 5.5 ft from surface to surface. Smaller spacings will cause temperatures to rise. If, for example, the casks are spaced 24 in. in one direction and 44 in. in the other, the temperature will increase by approximately 5 to 10 °F. This is due partly to the restricted radiant heat transfer to the sky and partly to the increase in ambient temperature at the interior of the array. All temperatures will still be within limits, but margins will decrease accordingly. The planned spacing for the 200 Area ISA is 4 ft in both directions. The temperature variation from 5.5-ft to 4-ft spacing was not reevaluated, as the tightened array described above had minimal effect. No temperature limits will be exceeded if the cask is positioned on its side due to an accidental drop or tipover. Although the ventilation flow will be ineffective, the situation will be no worse than the case of ash accumulation with plugged ducts, with temperatures remaining within the limits.

Neutron Radiography Facility TRIGA

The NRF TRIGA cask design pressure is 11.2 lb/in^2 gauge. The design pressure is based on the maximum pressure differential obtained from the reduced external pressure of 3.5 lb/in^2 absolute in Title 10, *Code of Federal Regulations*, Part 71, "Packaging and Transportation of Radioactive Material" (10 CFR 71), Section 10 CFR 71.71(c)(3). The DOT-6M/2R design pressure is 40 lb/in^2 gauge. The NRF TRIGA cask was pressure-tested at 17.3 lb/in^2 gauge, and the DOT-6M/2R container was pressure-tested at 60 lb/in^2 gauge. The pressure boundary for the DOT-6M/2R is the 2R container.

The low burnup of the SNF has produced a small quantity of fission products. Most of the fission product gases are captured within the fuel matrix. The fission product gases within each element (25 lb/in^2 maximum initial pressure) will result in negligible pressure in the cask or container (less than 2 lb/in^2), if all of the fuel elements are breached within the cask (WHC-SD-FF-TI-043). By-products from elastomeric seal degeneration will have minimal effect. Therefore, it is expected that the NRF TRIGA cask pressure will remain less than the design limit of 11.2 lb/in^2 gauge.

Light Water Reactor

Analysis determined the maximum canister pressures, due to fuel-rod rupture of worst-case hypothetical pressurized water reactor (PWR) and boiling water reactor (BWR) SNF assemblies for both normal and fire accident conditions. Analyses using the existing design of the NAC-1 cask (NAC-E-804) determined the maximum cask pressures due to fuel-rod rupture of a worst-case hypothetical PWR SNF assembly for both normal and fire accident conditions (SNF-4794). The pressure within the inner canister cavity due to rod rupture at maximum normal conditions for PWR fuel assemblies is 72.9 lb/in^2 absolute, which is 58.2 lb/in^2 gauge. This is within the design pressure of 75 lb/in^2 gauge established for the inner canister. The LWR canister was pressure tested during fabrication to 133% of the design pressure prior to loading. Worst-case canister pressures were also calculated for 100% rod rupture during the design basis fire conditions (SNF-4794) resulting in an inner canister cavity maximum average temperature of 540 °F. The resultant pressure at this temperature for the consolidated PWR fuel assemblies is 73.1 lb/in^2 gauge.

The consolidated BWR and PWR rod accident condition pressure was also calculated for both conditions. The resultant pressures of 21.2 lb./in^2 gauge and 28.5 lb/in^2 gauge are within the design pressure of the LWR canister and bounded by the fuel assembly pressures. There will be no failure of the LWR canister as a result of fuel rod rupture. The rod consolidation container has perforated end plates that preclude pressurization within the container.

D3.4.2.4.2 Source Term Analysis.

No source term is calculated because no release of radioactive material is identified for the conditions discussed in the subsections that follow.

Fast Flux Test Facility

The internal storage container pressure calculated for the release of all fission gases from the fuel rods is within the design pressure of the storage vessel.

Neutron Radiography Facility TRIGA

For TRIGA SNF, the internal storage container pressure calculated for the release of all fission gases from the fuel rods is within the design pressure of the storage vessel. Consequently, the casks will not release radioactive material.

Light Water Reactor Fuel

For the LWR fuel, the internal storage container pressure calculated for the release of all fission gases from the fuel rods is within the design pressure of the storage vessel.

D3.4.2.4.3 Consequence Analysis.

The postulated fuel rod rupture consequences have been prevented by the passive design features of the SNF casks. A consequence analysis is not needed.

D3.4.2.4.4 Comparison to Guidelines.

Unmitigated event – For events that exceed the initial conditions of the analyses, the bounding consequences for each fuel type show that the offsite release limits are not challenged. Therefore, no safety-class confinement SSCs are necessary.

For the FFTF fuel and the LWR fuel, the bounding onsite consequences for hypothetical releases outside of the initial conditions analyzed exceed the onsite risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

For the NRF TRIGA fuel, the unmitigated onsite consequences do not exceed the onsite risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following fuel rod rupture, no safety-significant SSCs are required to confine the NRF TRIGA fuel.

While no safety-class SSCs are required to provide confinement, some SSCs are required to protect the geometric configuration of the fuel for criticality safety. SSCs credited with retaining the critically safe geometry of the fuel are designated as safety class.

Mitigated event – The postulated accidents have been prevented by the passive design features of the casks; therefore, there is no release from this event.

D3.4.2.4.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls.

The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from heat loads, and keep the cask internal pressure within design values.

Fast Flux Test Facility Fuel

The following SSCs are designated as safety significant or safety class:

- **FFTF ISC** – The ISC is designated as safety significant to maintain confinement of the radioactive materials after the rupture of all fuel pins.
- **CCC inner container** – The CCC is designated as safety class to maintain structural integrity to provide criticality geometry control at the pressure generated from the rupture of all fuel pins.

The following TSR controls are necessary to protect the assumptions made in the cask internal pressure analyses:

- A minimum spacing array area shall be no less than 24 in. by 44 in. A method shall be established for indicating proper placement.
- There shall be no more than seven fuel assemblies in a CCC.

Neutron Radiography Facility TRIGA

Although the NRF TRIGA casks and DOT-6M inner 2R containers are not expected to leak as the result of fuel pin rupture, no safety-class or safety-significant SSCs are required to confine the potential release from the NRF TRIGA fuel based on the unmitigated analysis provided in Section D3.4.1.

Light Water Reactor Fuel

The following SSC is designated as safety significant and safety class:

- **LWR canister** – The LWR canister is designated as safety significant to maintain confinement of the radioactive materials and safety class to maintain structural integrity to provide criticality geometry control and a leaktight configuration at the pressure generated.

The following TSR control is necessary to protect the assumptions made in the cask pressure analysis:

- The number of individual fuel rods in the LWR canister does not exceed a maximum of 179 PWR rods or 96.5 BWR rods consolidated with 17 PWR rods.

D3.4.2.5 Seismic.

D3.4.2.5.1 Scenario Development.

A summary of the safety features required to prevent consequences after the seismic accident is provided in Table D3-15.

While the SNF storage systems are located at the 200 East Area, a design basis earthquake (DBE) could occur. The SNF storage systems could sustain seismically induced accelerations and forces, or impacts with nearby obstacles by overturning or sliding, which could potentially damage the structure of the storage system. These seismic accelerations and forces could affect the structural integrity or functionality of the SNF storage systems and potentially release radioactive materials out of the SNF storage system. All seismic *g* forces are well bounded by the drop accelerations.

Fast Flux Test Facility

200 Area ISA site-specific analyses in HNF-2183, *Overturning and Sliding Assessment for the Interim Storage Cask at the 200 East Area Interim Storage Area*, determined the margins of safety of the ISC in the 200 East Area against overturning and sliding during a seismic event. Other analyses were previously performed for a lower level earthquake than that currently considered for the 200 East Area (General Atomics 1995). Under the 200 East Area site-specific loadings (zero-period acceleration horizontal spectra of 0.26 *g* and vertical of 0.18 *g*, with a second earthquake of horizontal zero-period acceleration of 0.50 *g* and a vertical zero-period acceleration of two-thirds of the horizontal), the normal storage condition of the ISC was found to have a margin of safety against overturning of +0.20 (a factor of safety of +1.20) and against sliding of +0.33 (a factor of safety of +1.33). NRC criteria state that a minimum factor of safety of +1.1 must be maintained against sliding and overturning during a seismic event. These calculations found that the ISC had sufficient margins of stability during the DBE to not overturn and not slide.

Previous 400 Area ISA analyses (General Atomics 1995) also evaluated for overturning by simultaneously applying static loads to the ISC in the horizontal and vertical directions. The seismic motions from the DBE (0.25 *g*) earthquake were evaluated. These design response spectra were applied for the horizontal motion in the free field. For the vertical motion, the design response spectra were taken as two-thirds the response spectra for the horizontal motion. This analysis found that the ISC has a margin of safety of 0.29 against overturning during the DBE seismic event. Additionally, even if the seismic g level was higher and the ISC were to tip over, the tipover accident analysis performed for the ISC found that the cask still retains the stored SNF in a safe configuration.

Table D3-15. Summary of Safety Features Required to Withstand Seismic Accident.

Candidate accident	Checklist designator[a]	Safety function	Safety features (described in Chapter D4.0)
Natural phenomena - seismic event	R-01	Withstand seismic accelerations without loss of confinement, tipover, or sliding.	Safety class equipment for criticality geometry control: • CCC • LWR canister. Safety-significant equipment for confinement: • FFTF ISC • LWR canister. Safety-significant equipment for seismic stability/structural integrity: • FFTF ISC • ISO container • NAC-1[b] cask • ISO container. TSRs: • FFTF ISC and NAC-1[b] cask placed on concrete pad. • Rad-Vault[c] placed on compacted gravel. Defense-in-depth: • Rad-Vault[c] • NRF TRIGA[d] cask • DOT-6M[e] container • 2R container[f] • Canister Storage Building personnel are trained in sitewide and facility-specific emergency response.

[a] Checklist designators are from SNF-4820, 1999, *200 Area Interim Storage Area Final Hazard Analysis Report*, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.
[b] NAC-1 casks are manufactured by Nuclear Assurance Corporation.
[c] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.
[d] TRIGA is a trademark of General Dynamics Corporation.
[e] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.
[f] 2R designates the inner container of DOT-6M.

CCC = core component container.
FFTF = Fast Flux Test Facility.
ISC = interim storage cask.
ISO = International Standards Organization.
LWR = light water reactor.
NRF = Neutron Radiography Facility.
TSR = Technical Safety Requirement.

Movement of the ISC during an earthquake was evaluated. The analysis of the ISC for sliding shows that a coefficient of friction of at least 0.3 is required to prevent sliding motion. This result was compared to the coefficient of friction of concrete on concrete as determined from the *Manual of Concrete Practice* (ACI 1989, Part 4, Section 11.7.4.3). Since this value is 0.6, there is a large margin against sliding motion during an earthquake.

TRIGA

Using the current 200 East Area site-specific loadings, analyses were performed that demonstrate structural margins exist for the 200 Area ISA movement. The analyses (SNF-4792) found that the design of the Rad-Vault is well within the DBE for the site, and seismic stability is assured. The Rad-Vault will not overturn or slide on the soil. The lid lip will not fail, and the lid will remain on the Rad-Vault. The NRF TRIGA casks inside the container will not suffer loss of integrity, as their design for use in ground transport provides capacities far in excess of 0.26 *g* (WHC-SD-TP-SARP-008). Likewise, the DOT-6M containers that will be used for the FFCRs are qualified for conditions that are in excess of the 0.26 *g* ground acceleration earthquake analyzed for the 200 Area ISA (SNF-4792).

The margin of safety for overturning the loaded Rad-Vault is 1.1. Sliding was also examined using a friction factor of 0.35 between the gravel and the concrete base of the Rad-Vault. This factor was taken from the values shown in the *Uniform Building Code* (ICBO 1994) for sandy gravel and/or gravel. The margin of safety against sliding was 0.104.

The soil-bearing pressure beneath the Rad-Vault was not exceeded under seismic overturning loads. The margin of safety for soil bearing pressure was 0.32. As such, the Rad-Vault is not expected to overturn due to soil compaction loads.

Light Water Reactor

Current seismic criteria are specified in Table D2-2. Calculations were performed for the 200 Area ISA to evaluate the cask/container's resistance to overturning during the DBE (SNF-4794). The margin of safety against overturning the cask/container unit onto its left or right side is 2.7. The margin of safety against overturning the cask/container unit onto its front or back is 3.9. Based on these results, it was concluded that the cask/container units will not overturn.

Sliding of the cask/container was also examined using a friction factor of 0.5 between the concrete and the bottom side of the ISO shipping container. The margin of safety against sliding was found to be 0.58, which indicates that the cask/container will not slide.

Tipping of the storage system due to soil compaction beneath the pad under seismic overturning loads was also calculated for the 400 Area ISA. The margin of safety is 4.7.

D3.4.2.5.2 Source Term Analysis.

No source term is calculated because no release of radioactive material is identified.

D3.4.2.5.3 Consequence Analysis.

The consequences from the postulated seismic event are prevented by the passive design features of the SNF cask. A consequence analysis is not needed for the FFTF, TRIGA, or LWR fuel storage systems.

D3.4.2.5.4 Comparison to Guidelines.

Unmitigated event – For drops that exceed the initial conditions of the analyses, the bounding consequences for each fuel type show that the offsite release limits are not challenged. Drop analyses have been performed that would bound the seismic event. Therefore, no safety-class confinement SSCs are necessary.

For the NRF TRIGA fuel, the unmitigated onsite consequences do not exceed the onsite risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following a seismic event, no safety-significant SSCs are required to confine the NRF TRIGA fuel.

For the FFTF fuel and the LWR fuel in the NAC-1 casks, the bounding onsite consequences for a seismic event outside of the initial conditions analyzed exceed the onsite risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

Mitigated event – For the seismic events postulated and analyzed in the scenario development, the potential releases are prevented. These events take credit for the packaging components and limitations on handling (e.g., lift height) that have been shown to provide the impact protection and confinement of the fuel for the analyzed seismic events.

D3.4.2.5.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls.

The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from the seismic event within design values.

Fast Flux Test Facility Fuel

The following SSCs are designated as safety significant or safety class:

- **FFTF ISC** – The ISC is designated as safety significant to withstand seismic accelerations without loss of confinement or structural integrity, tipover or sliding, or to survive a tipover accident.
- **CCC** – The CCC is designated as safety class to withstand seismic accelerations without loss of structural integrity for geometry control.

The following TSR control is necessary to protect the assumptions made in the seismic analyses:

- The FFTF ISC shall be placed on a concrete pad.

Neutron Radiography Facility TRIGA

Although the NRF TRIGA casks and DOT-6M inner 2R containers are not expected to leak as the result of a seismic event, no safety-class or safety-significant SSCs are required to confine the potential release from the NRF TRIGA fuel based on the unmitigated analysis provided in Section D3.4.1.

The following TSR control is necessary to protect the assumptions made in the seismic event:

- The Rad-Vault shall be placed on compacted gravel.

Light Water Reactor Fuel

The following SSCs are designated as safety significant or safety class:

- **LWR canister** – The LWR canister is designated as safety class to provide a critically safe geometry for the SNF and retain a leaktight configuration. In addition, the LWR canister provides a safety significant confinement function for the SNF.
- **ISO container** – The ISO container is designated as safety significant for seismic stability and structural integrity protection.

The following TSR control is necessary to protect the assumptions made in the seismic event:

- The NAC-1 cask shall be placed on a concrete pad.

D3.4.2.6 Tornado/Wind.

D3.4.2.6.1 Scenario Development.

While the SNF storage systems are located at the 200 East Area, DBAs of straight wind and tornado wind could occur. Tornado missiles are not analyzed since probabilistic risk assessment analyses documented in HNF-1785, *Probabilistic Risk Analysis Tornado Missile Hazard to 200 Area Interim Storage Area*, have concluded that tornado missiles do not need to be considered. The SNF storage systems could sustain wind loadings or wind effects that could move the SNF storage systems to impact with nearby obstacles. Additionally, the SNF storage systems could suffer impacts from wind-driven missiles. These effects could affect the structural integrity or functionality of the SNF storage systems and potentially release radioactive materials out of the SNF storage system. A summary of the safety features required to prevent tornado and wind accident releases is provided in Table D3-16.

Fast Flux Test Facility

Analyses using the current 200 East Area site-specific requirements have been performed (SNF-4790). Calculations show that the loaded ISC will not tip over or slide. The increase in internal pressure due to differential pressure loading is negligible. Wind-driven missiles result in concrete penetration of less than 1 in.

Table D3-16. Summary of Safety Features Required to Prevent Tornado/Wind Accident.

Candidate accident	Checklist designator[a]	Safety function	Safety features (described in Chapter D4.0)
Natural phenomena - tornado and wind	R-06 R-08	Withstand DBT winds and the design basis wind and wind-driven missiles without loss of confinement. Without sliding or tip over.	Safety-significant equipment for confinement: • FFTF ISC • LWR canister. Safety-significant equipment for structural integrity: • FFTF ISC • NAC-1[b] cask • ISO container. TSRs: • FFTF ISC and NAC-1[b] cask placed on concrete pad. • Rad-Vault[c] placed on compacted gravel. Defense-in-depth: • Rad-Vault[c] • NRF TRIGA[d] cask • 2R container[e] • Operations personnel are trained in sitewide and facility-specific emergency response.

[a] Checklist designators are from SNF-4820, 1999, *200 Area Interim Storage Area Final Hazard Analysis Report*, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.
[b] NAC-1 casks are manufactured by Nuclear Assurance Corporation.
[c] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.
[d] TRIGA is a trademark of General Dynamics Corporation.
[e] 2R designates the inner container of DOT-6M. (DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation)

DBT = design basis tornado.
FFTF = Fast Flux Test Facility.
ISC = interim storage cask.
ISO = International Standards Organization.
LWR = light water reactor.
NRF = Neutron Radiography Facility.
TSR = Technical Safety Requirement.

TRIGA

Supplemental analyses using the current 200 East Area site-specific requirements have been performed demonstrating adequacy of the Rad-Vault to withstand the DBT and the wind-driven missile (SNF-4792).

Light Water Reactor

Analyses were performed for the 200 Area for a tornado wind speed of 200 mph and a differential pressure of 0.90 lb/in^2 over 3 seconds. The NAC-1 cask inside the ISO container does not slide or tip (SNF-4794). Wind missile damage does not reduce confinement or result in damage to the LWR canister or internal components.

D3.4.2.6.2 Source Term Analysis.

For spent fuel storage, the calculations determined that the design criteria were met and no releases are expected to occur as a result of this event. As such, source term analysis is not required.

D3.4.2.6.3 Consequence Analysis.

The consequences from the postulated tornado wind and wind-driven missile events are prevented by the passive design features of the robust SNF cask systems. A consequence analysis is not needed for the FFTF, TRIGA, or LWR fuel storage systems.

D3.4.2.6.4 Comparison to Guidelines.

Unmitigated event – For the bounding tornado/wind events, consequences for each fuel type show that the offsite release limits are not challenged. Therefore, no safety-class confinement SSCs are necessary.

For the NRF TRIGA fuel, the unmitigated onsite consequences do not exceed the onsite risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following bounding tornado/wind events, no safety-significant SSCs are required to confine the NRF TRIGA fuel.

For the FFTF fuel and the LWR fuel in NAC-1 casks, the bounding onsite consequences for the hypothetical accident outside of the initial conditions analyzed exceed the onsite risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

Mitigated event – For the tornado/wind events postulated and analyzed in the scenario development, the potential releases are prevented. These events take credit for the packaging components that have been shown to provide the impact protection and confinement of the fuel for the analyzed bounding tornado/wind events.

D3.4.2.6.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls.

The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from the bounding tornado/wind events within design values.

Fast Flux Test Facility Fuel

The following SSC is designated as safety significant:

- **FFTF ISC** – The ISC is designated as safety significant to withstand DBT winds (excluding DBT missiles) without sliding or tipover. The ISC must also withstand the design basis wind and wind-driven missiles established for the 200 Area ISA without loss of confinement, as well as provide structural integrity protection.

The following TSR control is necessary to protect the assumptions made in the tornado/wind event analyses:

- The FFTF ISC shall be placed on a concrete pad.

Neutron Radiography Facility TRIGA

No safety-class or safety-significant SSCs are required to confine the potential release from the NRF TRIGA fuel based on the unmitigated analysis provided in Section D3.4.1. The significant analytical margin for resistance to wind missile penetration of the thick Rad-Vault wall demonstrates that the general-service container is sufficient.

The following TSR control is necessary to protect the assumptions made in the tornado/ wind analysis:

- The Rad-Vault shall be placed on compacted gravel.

Light Water Reactor Fuel

The following SSCs are designated as safety significant:

- **ISO container** – The ISO container is designated as safety significant for structural integrity. Because external damage was assumed in the analysis, the ISO container side, end, and roof panels are not required for structural integrity and are therefore considered general service.

- **NAC-1 cask system** – The NAC-1 cask is designated as safety significant to withstand DBT winds (excluding DBT missiles) without sliding or tipover. The NAC-1 cask system must also withstand the design basis wind and wind-driven missiles established for the 200 Area ISA without structural damage to the inner canister.

- **LWR canister** – The LWR canister is designated as safety significant for confinement.

The following TSR control is necessary to protect the assumptions made in the tornado/wind events:

- The NAC-1 cask shall be placed on a concrete pad.

D3.4.2.7 Fire.

D3.4.2.7.1 Scenario Development.

A fire of unspecified origin can be assumed to occur, which subjects the exterior of the SNF storage containers to a high, short-term heat load that could ultimately threaten the confinement capability of the containers. A summary of the safety features required to prevent the design basis fire accident is provided in Table D3-17. The fire hazard analysis (SNF-4932) concluded that all fire scenarios are bounded by the transportation fire event. The transportation fire event is defined as an exposure of the exterior surface of the container system to a temperature of 1,475 °F for a period of 30 minutes.

Fast Flux Test Facility

In previous 400 Area ISA analyses (General Atomics 1995), the cask was evaluated for a hypothetical accident condition resulting from an external fire. In this transient case, the cask was subjected to a radiative environment of 1,475 °F with an emissivity of 0.9. After 30 minutes of exposure, the environment returned to normal ambient temperature, with no artificial cooling applied. The initial conditions for the transient were the steady-state temperatures for long-term normal conditions. In this accident case, the hot gas is assumed to enter the ventilation ducts and flow up the annulus between the liner and inner shield. The same flow model was used as for long-term normal conditions, except that the inlet temperature during the 30 minutes was taken to be 1475 °F instead of 94 °F. The fuel cavity average temperature calculated for this case was 261 °F, resulting in a cavity pressure that is below the design pressure, seal temperature limits and the cladding temperature limits are not exceeded. Therefore, a radioactive release will not occur.

Neutron Radiography Facility TRIGA

Previous 400 Area ISA analyses (WHC-SD-FF-TI-043) assumed that during a very hot summer, a fire (e.g., range fire, spilled fuel) engulfs the concrete Rad-Vault. The conditions of the fire are such that the outer wall of the Rad-Vault experiences a temperature of 1,475 °F for a period of 30 minutes. After 30 minutes of exposure, the environment returns to normal ambient temperature, without means of artificial cooling, and the Rad-Vault commences to cool down. An analysis for the 400 Area ISA was performed to determine the peak transient temperature profile of the vault wall, the peak temperature of the inner vault wall, and the fuel element cladding maximum temperature. This analysis showed a quasi steady-state temperature for the vault wall of 177 °F. Assuming equilibrium conditions, the inner vault wall temperature will result in a fuel cladding temperature of 193 °F, where the maximum allowable fuel cladding temperature is 302 °F. Therefore, radiological material releases will not result from exceeding the temperature limits; however, the design specifications for the concrete Rad-Vault may be exceeded for a short period of time at the exterior surface and should be evaluated if this event were to occur.

Table D3-17. Summary of Safety Features Required to Prevent Design Basis Fire Accident.

Candidate accident	Checklist designator[a]	Safety function	Safety features (described in Chapter D4.0)
Fire inside the ISA fence	L-03 L-07	Withstand fire conditions without loss of confinement or exceeding temperature limits for fuel cladding or cask components.	Safety class equipment for criticality geometry control: • CCC • LWR canister. Safety-significant equipment for confinement: • FFTF ISC • LWR canister. Safety-significant equipment for structural integrity: • NAC-1[b] cask • FFTF ISC • LWR canister. TSR: • Fire loadings are controlled per FHA. Defense-in-depth: • Rad-Vault[c] • NRF TRIGA[d] cask (inside Rad-Vault[c]) • DOT-6M[e] container • 2R container[f] (inside DOT-6M container and Rad-Vault[c]) • Canister Storage Building personnel are trained in sitewide and facility-specific emergency response.

[a] Checklist designators are from SNF-4820, 1999, *200 Area Interim Storage Area Final Hazard Analysis Report*, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.
[b] NAC-1 casks are manufactured by Nuclear Assurance Corporation.
[c] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.
[d] TRIGA is a trademark of General Dynamics Corporation.
[e] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.
[f] 2R designates the inner container of DOT-6M.

CCC = core component container.
FFTF = Fast Flux Test Facility.
FHA = fire hazards analysis.
ISA = interim storage area.
ISC = interim storage cask.
LWR = light water reactor.
NRF = Neutron Radiography Facility.
TSR = Technical Safety Requirement.

Light Water Reactor Fuel

The thermal analysis for the 400 Area ISA of the NAC-1 cask subjected to the transportation fire scenario of 10 CFR 71.73(c)(3) is documented in the original transportation SAR, *Safety Analysis Report for Nuclear Fuel Services, Inc., Spent Fuel Shipping Cask Model No. NFS-4* (NFS 1972). The original analyses assumed an internal thermal load of 11.5 kW, and the NAC-1 SAR assumed an internal 750 W thermal load. These analyses considered the NAC-1 cask with and without the ISO shipping container and assumed bounding conditions for the fire accident scenario, which has a maximum internal thermal load of 405 W. Examination of the previous fire accident has shown that these temperature limits are not exceeded for an internal thermal load of 11.5 kW or 750 W. The maximum temperature at the center of the fuel during a hypothetical fire accident is 589.8 °F (SNF-4794). The 644 °F temperature limit for the cladding is not exceeded anywhere in the array of stacked rods. The transportation safety analysis report for packaging (HNF-8705) and the NAC-1 SAR analyses found that the integrity of the NAC-1 cask was not lost for the design basis fire accident conditions and consequently, no credible radioactive releases occur for this event, as the LWR canister and internal components are not damaged.

D3.4.2.7.2 Source Term Analysis.

The SNF storage systems will not release radioactive material and fuel cladding temperature limits are not exceeded; therefore, a source term analysis is not required.

D3.4.2.7.3 Consequence Analysis.

Fast Flux Test Facility

The fire event does not result in a condition where the storage casks release radioactive material; therefore, a consequence analysis is not needed for this event.

Neutron Radiography Facility TRIGA

The fire event does not result in a condition where the storage casks release radioactive material; therefore, a consequence analysis has not been developed.

Light Water Reactor Fuel

The fire scenarios analyzed do not result in a scenario that causes the release of radioactive material; therefore, a consequence analysis has not been developed.

D3.4.2.7.4 Comparison to Guidelines.

Unmitigated event – For the hypothetical accidents described in Section D3.4.1, the bounding consequences for each fuel type show that the offsite release limits are not challenged. Therefore, no safety-class confinement SSCs are necessary.

For the NRF TRIGA fuel, the unmitigated onsite consequences do not exceed the onsite risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following a fire accident, no safety-significant SSCs are required to confine the NRF TRIGA fuel.

For FFTF fuel and the LWR fuel in NAC-1 casks, the bounding onsite consequences for the hypothetical accident analyzed exceed the onsite risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

While no safety-class SSCs are required to provide confinement, some SSCs are required to protect the geometric configuration of the fuel for criticality safety. SSCs credited with retaining the critically safe geometry of the fuel or a leaktight configuration to preclude water intrusion are designated as safety class. Temperatures from the design basis fire were evaluated to demonstrate the materials of construction can continue to perform their safety function.

Mitigated event – For the fire events postulated and analyzed in the scenario development, the potential releases are prevented. These events take credit for the packaging components providing confinement of the fuel for the analyzed fire events.

D3.4.2.7.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls.

The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from the fire events within design values.

The following TSR control is necessary to protect the assumptions made in the fire analyses:

- Fire loadings are to be controlled per the fire hazard analysis.

Fast Flux Test Facility Fuel

The following SSCs are designated as safety significant or safety class:

- **CCC** – The CCC is designated as safety class to maintain structural integrity for criticality geometry control.
- **FFTF ISC** – The ISC is designated as safety significant to withstand transportation design basis fire conditions without the loss of confinement or exceeding the temperature limits for fuel cladding or cask components, as well as to provide structural integrity protection.

Neutron Radiography Facility TRIGA

No safety-class or safety-significant SSCs are required to confine the potential release from the NRF TRIGA fuel based on the unmitigated analysis provided in Section D3.4.1. The significant analytical margin for resistance to fire temperatures based on thermal conductivity of the thick Rad-Vault wall demonstrates that the general-service container is sufficient. The Rad-Vault containing the NRF TRIGA casks and DOT-6M containers is not expected to exceed design basis temperature limits during the fire event.

Light Water Reactor Fuel

The following SSCs are designated as safety significant or safety class:

- **LWR canister** – The LWR canister is designated as safety class to maintain structural integrity for criticality geometry control and retain a leaktight configuration. In addition, the LWR canister is designated as safety significant for confinement.
- **NAC-1 cask system** – The NAC-1 cask is designated as safety significant to provide structural integrity protection against the fire event.

D3.4.3 Beyond Design Basis Accidents

DOE Order 5480.23 requires the evaluation of accidents beyond the design basis to provide a perspective of the residual risk associated with the operation of the facility. The beyond design basis accidents (BDBAs) are not analyzed to the same level of detail as DBAs, but do provide insight into the magnitude of consequences of BDBAs. This insight from BDBA analysis has the potential for identifying additional facility features that could prevent or reduce severe BDBA consequences. While these events would be beyond requirements of further safety-class or safety-significant functions, they might provide guidance to the prioritization of long-term safety improvements for a facility. The Order specifically excludes evaluation of human-generated external events as BDBAs.

Hypothetical release source terms are developed for each fuel type in Section D3.4.1.1 for an unmitigated release from each fuel type cask to determine the classification of engineered barriers that prevent an uncontrolled release. The scenarios are hypothetical and non-mechanistic. These hypothetical scenarios also establish the magnitude of a release from a BDBA.

Given the extremely low likelihoods associated with such BDBAs and the calculated consequences for the hypothetical scenarios in Section D3.4.1.1, no new insights into facility operations are expected.

D3.5 REFERENCES

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CHAPTER D4.0

SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS

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CONTENTS

CONTENTS (continued)

LIST OF TABLES

LIST OF TERMS

ACI	American Concrete Institute
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BWR	boiling water reactor
CCC	core component container
CSER	criticality safety evaluation report
DBA	design basis accident
DBE	design basis earthquake
DBF	design basis fire
DBT	design basis tornado
DFA	driver fuel assemblies
DOE	U.S. Department of Energy
FFCR	fuel follower control rod
FFTF	Fast Flux Test Facility
FSAR	final safety analysis report
GS	general service
IEM	interim examination and maintenance
ISA	interim storage area
ISC	interim storage cask
ISO	International Standards Organization
ITS	important to safety
LWR	light water reactor
NFS	Nuclear Fuel Services
NPH	natural phenomena hazard
NRC	U.S. Nuclear Regulatory Commission
NRF	Neutron Radiography Facility
PWR	pressurized water reactor
SAR	safety analysis report
SC	safety class
SNF	spent nuclear fuel
SS	safety significant
SSC	structure, system, and component
TSR	Technical Safety Requirement

D4.0 SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS

D4.1 INTRODUCTION

This chapter provides details of the structures, systems, and components (SSCs) that are necessary for the facility to meet offsite release limits, satisfy onsite risk guidelines, provide significant defense in depth, or contribute to worker safety. This chapter was written to the requirements of U.S. Department of Energy (DOE) Order 5480.23, *Nuclear Safety Analysis Reports*, follows the format guidance of DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, and complies with Title 10, *Code of Federal Regulations*, Part 830, "Nuclear Safety Management" (10 CFR 830). The chapter includes the following information:

- A list of safety-class SSCs and subsections containing safety function details, a system description, functional requirements, a system evaluation, and a list of assumptions requiring Technical Safety Requirements (TSRs) to ensure performance of the safety function.
- A list of the safety-significant SSCs and subsections containing safety function details, a system description, functional requirements, a system evaluation, and a list of assumptions requiring TSRs to ensure performance of the safety function.

In general, safety-class SSCs are those items required for protection of the offsite environment and the public, or the prevention of an inadvertent criticality. Safety-class SSCs include items designated as "safety class" in accordance with DOE Order 6430.1A, *General Design Criteria*. Safety-class SSCs also encompass items that are designated as "important to safety" and have been classified as Category A, as defined in Item 29 of HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements.*

Safety-significant items are those SSCs required for the protection of onsite personnel not directly involved in facility operations. Safety-significant SSCs may also encompass items that have been designated important-to-safety Category B in accordance with Item 29 of HNF-SD-SNF-DB-003, if they were not originally designed and procured in a U.S. Nuclear Regulatory Commission (NRC)-equivalent manner appropriate for their contents. SSCs that would prevent or mitigate an onsite fatality or protect large numbers of facility workers (not industrial safety), or those SSCs that would prevent or mitigate toxic chemical exposures are also designated safety significant.

All SSCs that are not classified as safety class or safety significant are general-service SSCs. The SSCs that have been designated as important-to-safety Category C are also included in general-service SSCs. General-service SSCs protect workers from standard industrial hazards or are controlled by Site safety programs.

The majority of the equipment in the 200 Area Interim Storage Area (ISA) is classified as general service. The dry cask storage systems for Fast Flux Test Facility (FFTF) and light water reactor (LWR) fuel have been classified as safety significant. The core component container (CCC) FFTF fuel and the LWR canister for commercial LWR fuel have been classified as safety class. This classification was based on NUREG/CR-6407, *Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety*, and DOE 6430.1A, Section 1300-4, because of their credited function for criticality prevention. The dry cask storage systems for TRIGA[1] fuel are classified general service, although they perform an NRC important-to-safety function. These items are currently in service at the 400 Area ISA. The general-service Neutron Radiography Facility (NRF) TRIGA casks and DOT-6M[2] containers were originally designed and procured in an NRC-equivalent manner such that application of the Spent Nuclear Fuel (SNF) Project NRC equivalency methodology imposing safety classification upgrades is not required. The risk of continued use of the general-service TRIGA fuel containers at the 200 Area ISA is considered acceptable due to the low-activity SNF, as discussed in the Section D3.4.1 analysis section.

The accident analyses in Chapter D3.0 identified one or more design basis accidents (DBAs) whose unmitigated consequences bound all accident sequences and scenarios identified for that category in the hazard analysis. The process identified criticality-related safety-class considerations requiring that the FFTF interim storage cask (ISC) CCC and the LWR canister provide criticality geometry control of the stored fuel. The analysis also identified safety-significant considerations for components of the FFTF and LWR cask storage systems that mitigated potential consequences in excess of onsite guidelines. No potential consequences in excess of onsite guidelines were identified for the TRIGA fuel. No events were identified that had offsite consequences in excess of allowable limits.

Each accident analysis section in Chapter D3.0 concludes with a summary of safety SSCs that provides the basis for this chapter. A summary of the accident categories and the designated safety SSCs that prevent or mitigate their consequences is provided in Table D4-1. Table D4-2 provides a summary of the safety SSCs identified and provides the level of safety credited for each accident category. Many SSCs provide a safety function for prevention or mitigation of more than one accident. Detailed definitions of safety-class and safety-significant SSCs are provided in Sections D4.3 and D4.4.

A probabilistic risk analysis was performed to determine the frequency of a tornado wind-borne missile striking any of the dry cask storage systems planned for the 200 Area ISA. The analysis, which is documented in HNF-1785, *Probabilistic Risk Analysis Tornado Missile Hazard to 200 Area Interim Storage Area*, concludes that the rate at which storage area safety is compromised is less than 10^{-8} per year. Generation of tornado-based, wind-borne missiles that could compromise cask system integrity is therefore not credible and evaluation is not required.

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[1] TRIGA is a trademark of General Dynamics Corporation.

[2] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.

Table D4-1. Safety-Related Structures, Systems, and Components Summary. (7 sheets)

Design Basis Accident (Chapter D3.0)	Safety structures, systems and components[a]		Safety function	Summary justification of safety function	Controls
	Description	Designation			
Handling/Drop	FFTF Interim Storage Cask	SS	Maintain confinement of radioactive materials after a credible drop and provide passive protection of the CCC such that it retains structural integrity.	Design requirement of the ISC is to maintain confinement after credible drops. The CCC inside the ISC provides support to the fuel assemblies and maintains geometry control for criticality prevention. Structural analysis demonstrates integrity of the CCC.	AC 5.9 Lift Restrictions – lift height restriction.
	CCC	SC	Maintain structural integrity within the ISC to provide criticality geometry control after credible drops.	The CCC inside the ISC provides support to the fuel assemblies and maintains geometry control for criticality prevention. Structural analysis, including the buckling analysis, demonstrates integrity of the CCC.	AC 5.9 Lift Restrictions – lift height restriction.
	NRF TRIGA[b] Cask	GS[c]	Maintain confinement of radioactive materials after a credible drop.	Design requirement of the cask is to maintain confinement after credible drops.	AC 5.9 Lift Restrictions – lift height restriction.
	DOT-6M[d] Container	GS[c]	Provide impact absorption for the 2R[e] container.	Design requirement of the DOT-6M[d] is to provide impact absorption so the 2R[e] container can maintain confinement after credible drops.	AC 5.9 Lift Restrictions – lift height restriction.
	2R[e] Container	GS[c]	Maintain confinement of radioactive materials after a credible drop within the DOT-6M.[d]	Design requirement of the 2R[e] container is to maintain confinement after credible drops.	AC 5.9 Lift Restrictions – lift height restriction.
	NAC-1[f] Cask	SS	Provide passive protection of the LWR canister such that it retains structural integrity after a credible drop.	Design requirement of the NAC-1[f] cask is to provide impact absorption so the LWR canister can maintain confinement after credible drops.	AC 5.9 Lift Restrictions – lift height restriction.
	LWR Canister	SC/SS	Maintain confinement of radioactive materials after a credible drop (SS) within NAC-1[f] cask and maintain structural integrity (including internal components) to provide criticality geometry control (SC).	The LWR canister maintains confinement after credible drops. The canister design provides support to the fuel assembly or pins and maintains geometry control for criticality prevention. Leak tight configuration precludes water intrusion precluding criticality. Structural analysis including buckling analysis demonstrates integrity of the LWR canister.	AC 5.9 Lift Restrictions – lift height restriction.

Table D4-1. Safety-Related Structures, Systems, and Components Summary. (7 sheets)

Design Basis Accident (Chapter D3.0)	Safety structures, systems and components[a]		Safety function	Summary justification of safety function	Controls
	Description	Designation			
Mobile Crane Fall	FFTF Interim Storage Cask	SS	Maintain structural integrity sufficient to maintain confinement of radioactive materials after the crane fall.	The ISC is analyzed to maintain confinement after a crane fall accident.	AC 5.11 Crane Utilization.
	NAC-1[f] Cask System	SS	Provide passive protection of the LWR canister such that it maintains confinement after a crane fall accident.	The NAC-1[f] cask is analyzed to provide impact absorption so the LWR canister can maintain confinement after credible drops.	AC 5.11 Crane Utilization.
Cask Tipover	FFTF Interim Storage Cask System	SS	Maintain confinement of radioactive materials after cask tipover and provide passive protection of the CCC such that it retains structural integrity.	Design requirement of the ISC is to maintain confinement after cask tipover. The CCC inside the ISC provides support to the fuel assemblies and maintains geometry control for criticality prevention. Structural analysis demonstrates integrity of the CCC.	AC 5.10 Spacing and Placement.
	CCC	SC	Maintain structural integrity to provide criticality geometry control.	The CCC inside the ISC provides support to the fuel assemblies and maintains geometry control for criticality prevention. Structural analysis, including the buckling analysis, demonstrates integrity of the CCC.	(Design feature)
	Rad-Vault[g]	GS[c]	Passive design features preclude tipping.	The Rad-Vault[g] has been analyzed to not tip over or slide as a result of NPH or DBAs.	AC 5.10 Spacing and Placement. (Design feature)
	NRF TRIGA[b] Cask	GS[c]	Maintain confinement of radioactive materials after cask tipover.	Drop analyses bound tipover forces and demonstrate confinement is maintained.	(Design feature)
	2R[e] Container in DOT-6M	GS[c]	Maintain confinement of radioactive materials after container tipover.	Drop analyses bound tipover forces and demonstrate confinement is maintained.	(Design feature)
	NAC-1[f] Cask in ISO Container	SS	Passive design features preclude tipping.	The NAC-1[f] cask in the ISO container has been analyzed to not tip over or slide as a result of NPH or DBAs.	AC 5.10 Spacing and Placement. (Design feature)

Table D4-1. Safety-Related Structures, Systems, and Components Summary. (7 sheets)

Design Basis Accident (Chapter D3.0)	Safety structures, systems and components[a]		Safety function	Summary justification of safety function	Controls
	Description	Designation			
Fuel Rod Rupture	FFTF Interim Storage Cask	SS	Maintain confinement of radioactive materials after rupture of all fuel pins.	Design requirement of the ISC is to maintain confinement after rupture of all fuel pins by not exceeding the design pressure.	AC 5.8 Source Inventory Receipt Inspection. AC 5.10 Spacing and Placement.
	CCC	SC	Maintain structural integrity to provide criticality geometry control at the pressure generated from the rupture of all fuel pins.	Design requirement of the CCC is to structurally withstand the pressure after rupture of all fuel pins by not exceeding the design pressure.	(Design feature)
	NRF TRIGA[b] Cask	GS[c]	Maintain confinement of radioactive materials after rupture of all fuel pins.	Design requirement of the TRIGA[b] cask is to maintain confinement after rupture of all fuel pins by not exceeding the design pressure.	(Design feature)
	2R[e] Container	GS[c]	Maintain confinement of radioactive materials after rupture of all fuel pins.	Design requirement of the 2R[e] container is to maintain confinement after rupture of the fuel pin by not exceeding the design pressure.	(Design feature)
	LWR Canister	SC/SS	Maintain confinement of radioactive materials (SS) and maintain structural integrity to provide criticality geometry control (SC) at the pressure generated.	Design requirement of the LWR canister is to maintain confinement and structurally withstand the pressure after rupture of all the fuel pins by not exceeding the design pressure.	AC 5.8 Source Inventory Receipt Inspection.
Seismic	FFTF Interim Storage Cask	SS	Withstand seismic accelerations without loss of structural integrity sufficient to maintain confinement, and without tipover or sliding.	Although the ISC was analyzed to survive a cask tipover, seismic analysis precludes common mode failure involving multiple casks.	AC 5.10 Spacing and Placement.
	CCC	SC	Withstand seismic accelerations to maintain structural integrity to provide criticality geometry control.	All seismic accelerations are bounded by the drop analysis.	(Design feature)
	Rad-Vault[g]	GS[c]	Withstand seismic accelerations without tipover or sliding.	Seismic analysis precludes common mode failure involving multiple casks and supplants requirement to perform cask tipover analysis.	AC 5.10 Spacing and Placement.

Table D4-1. Safety-Related Structures, Systems, and Components Summary. (7 sheets)

Design Basis Accident (Chapter D3.0)	Safety structures, systems and components[a]		Safety function	Summary justification of safety function	Controls
	Description	Designation			
Seismic (continued)	NRF TRIGA[b] Cask	GS[c]	Withstand seismic accelerations without loss of confinement.	All seismic accelerations are bounded by drop analysis.	(Design feature)
	DOT-6M[d]	GS[c]	Provide impact absorption for the 2R[e] container.	Design requirement of the DOT-6M[d] is to provide impact absorption so the 2R[e] container can maintain confinement after credible drops.	(Design feature)
	2R[e] Container	GS[c]	Withstand seismic accelerations without loss of confinement.	Seismic accelerations are bounded by the drop analysis.	(Design feature)
	LWR Canister	SC/SS	Withstand seismic accelerations without loss of confinement (SS) and maintain structural integrity (including internal components) to provide criticality geometry control (SC).	All seismic accelerations are bounded by drop analysis.	(Design feature)
	NAC-1[f] Cask in ISO Container	SS	Provide structural support to the LWR canister to withstand seismic accelerations without tipover or sliding.	The NAC-1[f] cask in the ISO container has been analyzed not to tip over or slide as a result of a seismic event.	AC 5.10 Spacing and Placement. (Design feature)
Tornado/Wind	FFTF Interim Storage Cask	SS	Withstand DBT winds (excluding DBT missiles) without sliding or tipover. Also withstand the design basis wind and wind-driven missiles established for the 200 Area ISA without loss of structural integrity sufficient to cause loss of confinement.	Tornado analysis precludes common mode failure involving multiple casks, and the wind missile analysis demonstrates no loss of confinement and environmental protection provided to the CCC.	AC 5.10 Spacing and Placement.
	Rad-Vault[g]	GS[c]	Withstand DBT winds (excluding DBT missiles) without sliding or tipover. Also withstand the design basis wind and wind-driven missiles established for the 200 Area ISA.	Tornado analysis precludes common mode failure involving multiple casks, and the wind missile analysis demonstrates environmental protection provided to TRIGA[b] casks and DOT-6M[d]/2R[e] containers.	AC 5.10 Spacing and Placement.
	NRF TRIGA[b] Cask	GS[c]	Withstand tornado pressure differential without loss of confinement.	The design pressure is much greater than the 0.9 lb/in^2 gauge tornado pressure differential.	(Design feature)

Table D4-1. Safety-Related Structures, Systems, and Components Summary. (7 sheets)

Design Basis Accident (Chapter D3.0)	Safety structures, systems and components[a]		Safety function	Summary justification of safety function	Controls
	Description	Designation			
Tornado/Wind (continued)	2R[e] Container	GS[c]	Withstand tornado pressure differential without loss of confinement.	The design pressure is much greater than the 0.9 lb/in^2 gauge tornado pressure differential.	(Design feature)
	NAC-1[f] Cask	SS	Withstand DBT winds (excluding DBT missiles). Also withstand the design basis wind and wind-driven missiles established for the 200 Area ISA, without structural damage to the LWR canister.	Tornado analysis precludes common mode failure involving tipover of multiple casks, and the wind-missile analysis demonstrates environmental protection provided to the LWR canister.	AC 5.10 Spacing and Placement.
	LWR Canister	SS	Withstand tornado pressure differential without loss of confinement.	The design pressure is much greater than the 0.9 lb/in^2 gauge tornado pressure differential.	(Design feature)
	ISO Container	SS	Provide structural support to the NAC-1[f] cask to withstand tornado winds without tipover or sliding.	The NAC-1[f] cask in the ISO container has been analyzed not to tip over or slide as a result of a tornado event.	AC 5.10 Spacing and Placement.
Fire	FFTF Interim Storage Cask	SS	Withstand transportation DBF conditions without loss of structural integrity sufficient to cause loss of confinement or exceeding temperature limits for fuel cladding or cask components.	ISC design precludes exceeding temperature limits for fuel cladding or cask components and maintains confinement during a DBF. Precludes common mode failure involving multiple casks.	AC 5.12 Combustible Loading Limits.
	CCC	SC	Withstand transportation DBF conditions inside ISC without exceeding temperature limits for fuel cladding or container components to maintain structural integrity for criticality geometry control.	CCC can withstand temperatures inside ISC during a DBF such that temperatures do not exceed limits, maintaining structural integrity.	AC 5.12 Combustible Loading Limits.
	Rad-Vault[g]	GS[c]	Withstand transportation DBF conditions such that TRIGA[b] cask and DOT-6M[d]/2R[e] containers inside do not lose confinement, or exceed temperature limits for fuel cladding or cask components.	Rad-Vault[g] provides environmental protection of the TRIGA[b] casks and DOT-6M[d]/2R[e] containers during a DBF such that their confinement function is not breached and temperatures do not exceed limits. Precludes common mode failure involving multiple casks.	AC 5.12 Combustible Loading Limits.

Table D4-1. Safety-Related Structures, Systems, and Components Summary. (7 sheets)

Design Basis Accident (Chapter D3.0)	Safety structures, systems and components[a]		Safety function	Summary justification of safety function	Controls
	Description	Designation			
Fire (continued)	NRF TRIGA[b] Cask	GS[c]	Withstand transportation DBF conditions within the Rad-Vault[g] without losing confinement or exceeding temperature limits for fuel cladding or cask components.	TRIGA[b] casks can withstand temperatures inside the Rad-Vault[g] during a DBF such that their confinement function is not breached and temperatures do not exceed limits.	AC 5.12 Combustible Loading Limits.
	DOT-6M[d]	GS[c]	Withstand transportation DBF conditions within the Rad-Vault[g] such that the 2R[e] container inside does not lose confinement or exceed temperature limits for fuel cladding or container components.	DOT-6M[d] containers provide thermal insulation during a DBF such that the 2R[e] container inside does not lose its confinement function and temperatures do not exceed limits.	AC 5.12 Combustible Loading Limits.
	2R[e] Container	GS[c]	Withstand transportation DBF conditions within the Rad-Vault[g] without losing confinement or exceeding temperature limits for fuel cladding or container components.	2R[e] containers can withstand temperatures inside the Rad-Vault[g] during a DBF such that their confinement function is not breached and temperatures do not exceed limits.	AC 5.12 Combustible Loading Limits.
	NAC-1[f] Cask	SS	Withstand transportation DBF conditions such that the LWR canister does not lose confinement or exceed temperature limits for fuel cladding or cask components.	NAC-1[f] cask provides environmental protection of the LWR canisters during a DBF such that their confinement function is not breached and temperatures do not exceed limits. Precludes common mode failure involving multiple casks.	AC 5.12 Combustible Loading Limits.
	LWR Canister	SC/SS	Withstand transportation DBF conditions inside the NAC-1[f] cask without loss of confinement (SS) or exceeding temperature limits for fuel cladding or container components to maintain structural integrity for criticality geometry control (SC).	LWR canisters can withstand temperatures inside the NAC-1[f] cask during a DBF such that confinement is not breached and temperatures do not exceed limits, maintaining structural integrity and a leaktight configuration.	AC 5.12 Combustible Loading Limits.

Table D4-1. Safety-Related Structures, Systems, and Components Summary. (7 sheets)

Design Basis Accident (Chapter D3.0)	Safety structures, systems and components[a]		Safety function	Summary justification of safety function	Controls
	Description	Designation			
Criticality	FFTF CCC	SC	Maintain structural integrity to provide criticality geometry control under all credible normal and accident conditions using double contingency principles and $K_{eff} \leq 0.95$.	Dimensional parameters credited in the CSERs must be maintained under all credible normal and accident conditions by retaining structural integrity of credited components, as demonstrated in the structural analyses, including the buckling analysis.	AC 5.7 Nuclear Criticality Safety. AC 5.9 Lift Restrictions.
	LWR Canister	SC	Maintain structural integrity (including internal components) to provide criticality geometry control under all credible normal and accident conditions using double contingency principles and $K_{eff} \leq 0.95$. Maintain leaktight configuration to prevent water intrusion.	Dimensional parameters credited in the CSERs must be maintained under all credible normal and accident conditions by retaining structural integrity of credited components, as demonstrated in the structural analyses, including the buckling analysis. The LWR canister must remain leaktight to prevent water intrusion, which precludes the possibility of criticality even if damaged fuel is within the canister.	AC 5.7 Nuclear Criticality Safety. AC 5.9 Lift Restrictions.

[a] Safety function identifies safety-significant or safety-class function required to prevent or mitigate the specific accident.
[b] TRIGA is a trademark of General Dynamics Corporation.
[c] NRC important-to-safety Category B function.
[d] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.
[e] 2R designates the inner container of DOT-6M.
[f] NAC-1 casks are manufactured by Nuclear Assurance Corporation.
[g] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.

CCC = core component container.
CSER = criticality safety evaluation report.
DBA = design basis accident.
DBF = design basis fire.
DBT = design basis tornado.
FFTF = Fast Flux Test Facility.
GS = general service.
ISA = interim storage area.
ISC = interim storage cask.
ISO = International Standards Organization.
LWR = light water reactor.
NPH = natural phenomena hazard.
NRF = Neutron Radiography Facility.
SC = safety class.
SS = safety significant.

Table D4-2. Structure, System, and Component Safety Function Classification Summary.

Structures, systems, and components	Handling/ drop accident	Mobile crane fall	Cask tipover	Fuel rod rupture	Seismic	Tornado/ wind	Fire	Confinement boundary	Criticality prevention	NRC important to safety classification	FSAR section
FFTF ISC	SS	SS	SS	SS	SS	SS	SS	X		ITS-Cat B	D4.4.1
CCC	SC[a]		SC[a]	SC[a]	SC[a]		SC[a]		X	ITS-Cat A	D4.3.1
Rad-Vault[b]			GS[c]		GS[c]	GS[c]	GS[c]			ITS-Cat B	D4.4.2
NRF TRIGA[d] Cask	GS[c]		GS[c]	GS[c]	GS[c]	GS[c]	GS[c]	X		ITS-Cat B	D4.4.3
DOT-6M[e]	GS[c]		GS[c]		GS[c]		GS[c]			ITS-Cat B	D4.4.4
2R[f] Container	GS[c]		GS[c]	GS[c]	GS[c]	GS[c]	GS[c]	X		ITS-Cat B	D4.4.4
NAC-1[g]	SS	SS	SS		SS	SS	SS			ITS-Cat B	D4.4.5
LWR Canister[h]	SS/SC[a]			SS/SC[a]	SS/SC[a]	SS	SS/SC[a]	X	X	ITS-Cat A	D4.3.2
ISO Container			SS		SS	SS				ITS-Cat B	D4.4.5
Crane	GS[i]	GS[i]								ITS-Cat B	Not applicable
Lifting rigging	GS[i]									ITS-Cat B	Not applicable

[a] These items are designated as safety class for structural integrity to provide criticality geometry control or leaktight configuration to preclude water intrusion.
[b] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.
[c] NRC important-to-safety Category B function.
[d] TRIGA is a trademark of General Dynamics Corporation.
[e] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.
[f] 2R designates the inner container of DOT-6M.
[g] NAC-1 casks are manufactured by Nuclear Assurance Corporation.
[h] Includes internal components, support structures, baskets, and rod consolidation assembly.
[i] Critical lift requirements imposed using DOE/RL-92-36, 1993, *Hanford Site Hoisting and Rigging Manual*, accomplish NRC equivalency for important-to-safety Category B.

CCC	=	core component container.
FFTF	=	Fast Flux Test Facility.
FSAR	=	final safety analysis report.
GS	=	general service.
ISC	=	interim storage cask.
ISO	=	International Standards Organization.
ITS-Cat	=	important to safety category.
LWR	=	light water reactor.
NRC	=	U.S. Nuclear Regulatory Commission.
NRF	=	Neutron Radiography Facility.
SC	=	safety class.
SS	=	safety significant.

D4.2 REQUIREMENTS

This section identifies design codes, standards, regulations, and DOE Orders that are required for establishing the facility safety basis. The intent is to provide only the requirements that are specific to this chapter and pertinent to the safety basis. Specific codes, standards, and requirements applicable to the 200 Area ISA are defined in HNF-SD-SNF-RD-001, *Spent Nuclear Fuel Project Standards/Requirements Identification Document.*

The following DOE Orders, regulations, and standards are applicable to the safety basis for the facility:

- 10 CFR 830, Subpart A, "Quality Assurance Requirements." This rule requires that a sufficient quality assurance program be in place.
- Title 10, *Code of Federal Regulations*, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR 72). This rule is used for the licensing of independent spent fuel storage installations. 10 CFR 72.3, "Definitions," defines SSCs that are considered "important to safety."
- Title 10, *Code of Federal Regulations*, Part 835, "Occupational Radiation Protection" (10 CFR 835). This rule provides requirements for radiation protection programs.
- DOE Order 5480.22, *Technical Safety Requirements*. This order sets the requirements for the development and preparation of a TSR document, which is prepared separately.
- DOE Order 5480.23, *Nuclear Safety Analysis Reports*. This order provides nuclear safety analysis report (SAR) content requirements.
- DOE Order 5480.28, *Natural Phenomena Hazards Mitigation*. This order is used to define design requirements for seismic events and straight wind.
- DOE Order 6430.1A, *General Design Criteria*. This order provides general criteria and guidance for facility and system design. In addition, Division 13, "Special Facilities," Section 1300, "General Requirements," and Section 1320, "Irradiated Fissile Material Storage Facilities," requirements are imposed for the 200 Area ISA. Compliance with DOE Order 6430.1A is demonstrated in SNF-5139, *200 Area Interim Storage Area DOE 6430.1A Compliance Evaluation.*
- DOE/RL-92-36, *Hanford Site Hoisting and Rigging Manual.* This manual provides the requirements for lifting and rigging equipment and material.

The following standards are used for format and content guidance:

- DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports.* This standard is used to determine hazard categories for nuclear facilities.

- DOE-STD-1020-94, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*. This standard provides natural phenomena hazard (NPH) design and evaluation criteria. The 200 Area ISA was designed and evaluated for seismic events and straight wind in accordance with this standard.

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

- DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*. This standard supplements DOE Order 5480.23 by providing guidance specific to nonreactor nuclear facilities. In this regard, the standard provides more detailed information on the performance of accident analyses for Hazard Category 2 and 3 facilities. The standard also establishes additional requirements for the establishment of defense in depth and the identification of safety-significant SSCs.

In Letter 9504163/95-SFD-167, *Implementation of K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy* (Sellers 1995), DOE established the requirement for new SNF Project facilities to achieve "nuclear safety equivalency" to comparable NRC-licensed facilities. The SNF Project identified the NRC requirements that were needed in addition to existing applicable DOE requirements to establish nuclear safety equivalency. These NRC requirements and the process used to identify them are documented in HNF-SD-SNF-DB-003. Applicable requirements have been imposed for the ISA. The NRC requirements are used for licensing independent spent fuel storage installations. 10 CFR 72.3, "Definitions," defines SSCs that are considered important to safety. 10 CFR 72.122, "Overall Requirements," requires that the design bases for SSCs important-to-safety reflect appropriate combinations of effects of normal and accident conditions and the effects of natural phenomena.

For the 200 Area ISA, important-to-safety SSCs have been identified in accordance with 10 CFR 72.3. Once SSCs have been identified as having a function meeting the definition of important to safety, the requirements for SSCs important to safety specified in 10 CFR 72 are imposed. A graded approach is applied to an SSC important to safety by using the guidance provided in NUREG/CR-6407, as follows:

- Category A – Critical to Safe Operation

  SSCs in this category include those whose failure or malfunction could directly result in a condition adverse to public health and safety. Important-to-safety SSCs in this category are classified as safety class, as defined in DOE Order 6430.1A with the additional requirements therein.

- Category B – Major Impact on Safety

  SSCs in this category include those whose failure or malfunction could result in a condition adversely affecting collocated worker health and safety. Note that from the definition of Category C, Category B is understood to include events that could significantly damage the storage containers without severe impact to public health and safety. SSCs in this category are classified as safety significant if they were not originally designed and procured in an NRC-equivalent manner appropriate for their contents.

- Category C – Minor Impact on Safety

  SSCs whose failure or malfunction would not significantly reduce containment and would not be likely to create a situation adversely affecting public or collocated workers' health and safety. SSCs in this category are classified as general service.

In conjunction with the requirements noted above, ANSI/ANS-57.9-1992, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)*, provides guidance on design and performance criteria specific to a dry storage concept. These criteria are intended to be consistent with meeting the 10 CFR 72 requirements.

Additional discussion of the NRC criteria for mitigation of natural phenomena is provided in Section B1.2 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 3, "Annex B – Cold Vacuum Drying Facility Final Safety Analysis Report." Specific NPH design requirements implementing NRC equivalency requirements have been established for the 200 ISA via Engineering Change Notice 643545 to WHC-SD-SNF-DB-009, *Canister Storage Building Natural Phenomena Hazards*, which added Appendix C, "200 East Area Interim Storage Area Natural Phenomena Hazards."

Letter 9757710/97-SFD-172, *Contract No. DE-AC06-96RL13200 – Risk Evaluation Guidelines (REGs) to Ensure Inherently Safer Designs* (Sellers 1997), provides SNF Project guidance for design criteria with regard to exposure limits to be used in evaluating risk to the public and to onsite personnel.

These documents establish the design requirements for the 200 Area ISA.

D4.3 SAFETY-CLASS STRUCTURES, SYSTEMS, AND COMPONENTS

Safety-class items are SSCs, including portions of storage systems, whose failure could adversely affect the environment or the safety and health of the public. Detailed definitions of safety-class SSCs are provided in Section D4.1.

The SSCs credited with a safety-class function in Chapter D3.0 are described in the subsections that follow.

D4.3.1 Core Component Container for Fast Flux Test Facility Spent Fuel

The CCC is designated safety class for criticality geometry control only.

D4.3.1.1 Core Component Container Safety Function.

The CCC is designated safety class for criticality geometry control, which is the only safety-class function identified for this SSC. The criticality safety evaluation summarized in Chapter D6.0 demonstrates that criticality is not credible for storage of fuel assemblies within the CCC/ISC. The assumption of full water flooding is considered to be not credible and the fixed geometry of the closely packed, CCC, seven-tube arrangement physically restrains the assemblies/pin containers such that criticality is not possible.

The DBAs identified in Chapter D3.0 that could impose structural loads or accelerations on the CCC inside the ISC have the potential to change the criticality geometry control structure (Table D4-1 and Table D4-2). Criticality safety evaluation reports (CSERs) discussed in Chapter D6.0 credit the CCC with providing the structure that maintains the critically safe geometry. The functional requirements and evaluations for the CCC (Sections D4.3.1.3 and D4.3.1.4) include summary discussions of the bounding DBAs using the following analyses:

- Handling/drop – Maintain structural integrity within the ISC to provide criticality geometry control after credible drops.
- Cask tipover – Maintain structural integrity to provide criticality geometry control.
- Fuel rod rupture – Maintain structural integrity to provide criticality geometry control up to and including the pressure generated from the rupture of all fuel pins.
- Seismic – Withstand seismic accelerations to maintain structural integrity for criticality geometry control.
- Fire – Withstand transportation design basis fire (DBF) conditions inside the ISC, without exceeding temperature limits for fuel cladding or container components, to maintain structural integrity for criticality geometry control.

D4.3.1.2 Core Component Container Description.

The CCC (Figure D2-9) is an unshielded, sealed fuel storage container with seven fuel storage positions. As a defense-in-depth feature, the CCC provides canning for the FFTF fuel during the spent fuel dry storage 40-year design lifetime. The CCC also provides the geometry to ensure criticality control during handling and storage of the fuel. The CCC is designed such that it is fully retrievable from the storage configuration, although the capability to remove individual fuel components from a CCC is not required nor is it guaranteed. The center storage location can accept a fuel assembly that has had the bottom 15.5 in. removed. However, due to the indented CCC grapple-handling socket, an Ident-69 pin container cannot be stored in the center location. Therefore, the maximum loading of a CCC will be either six Ident-69 pin containers or seven FFTF fuel assemblies. Pin containers and fuel assemblies have also been analyzed for a mixture of up to five Ident-69 containers and two driver fuel assemblies (DFAs) in a CCC. The Ident-69 container is allowed to contain a maximum of 217 pins, the same number as in a DFA.

The CCC was designed and fabricated in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (ASME Code), Section VIII, Division 2 (ASME 1989). The CCC is fabricated from stainless steel and nickel alloy material (approved American Society for Testing and Materials [ASTM] materials for both ASME VIII and ASME III) to provide corrosion-resistant fuel storage, with overall dimensions of 20.0 in. in diameter by 146 in. in height. The weight of the empty CCC is 1,100 lb. The maximum weight of a loaded CCC occurs with seven DFAs. The weight of the seven DFAs is 3,900 lb, giving the CCC a maximum gross weight of 5,000 lb. The storage positions of the CCC are formed by seven steel tubes arranged with six equally spaced, radial storage positions surrounding the seventh center storage position. There is minimal clearance between storage tubes in the bundle orientation. The materials used in the fabrication of the CCC are primarily ASTM-certified 304 stainless steels except for the lower approximately 12-in. portions of the tubes and the lower cups, which are ASTM-certified nickel alloy material UNS06625, and the closure bolts, which are certified to ASTM A-574 in accordance with the *Annual Book of ASTM Standards* (ASTM 1989). The bottom cups provide extra corrosion resistance for each cleaned and dried fuel assembly. These cups are free to move axially (0.12 in.) within the lower support plate to accommodate thermal expansion. The lower support plate provides guidance and spacing for each of the six tube assemblies. The reducing section (saddle) maintains the outside envelope of the container and supports the fuel assemblies.

The upper portion of the outer tubes has a 6.69 in. outside diameter, with 109-mil thick walls. There is a saddle section 14 in. above the bottom of the CCC, where each outer storage tube transitions to a smaller section measuring 4.0 in. in outside diameter, with 226-mil thick walls. The bottom 12.4 in. of the lower section is fabricated from nickel alloy material for enhanced corrosion resistance. The center tube has a 6.54 in. outside diameter, with 120-mil thick walls. The bottom 10.0 in. of the center tube is also fabricated from nickel alloy material. There is no size reduction at the lower end of the center tube.

The outer storage tubes are suspended from the upper support plate. The center storage tube connects the upper support plate with the lower support plate. The outer tubes are fixed only at their upper ends so they are free to accommodate thermal expansion. This design also permits the outer tubes to stretch slightly to absorb energy during a CCC drop accident onto the ISC internal impact limiter. Drop energy is absorbed by the CCC tubes until the gap between the tube stop and the lower support plate is taken up. The CCC then acts as a rigid body for final interface with the ISC impact limiter during an accident.

The lower support plate has an 18.0-in. diameter and is 1.5 in. thick. It limits downward travel of the outer storage tubes and also provides radial positioning guidance for inserting the CCC into the ISC. The upper support plate has a 20.0-in. diameter, with an overall thickness of 3.6 in. It provides support for the outer storage tubes and the seating surface for the cover seal. There are twelve drilled holes in the upper support plate that accommodate the cover bolts.

The cover has a 20.0-in. diameter and an overall thickness of 1.63 in. The lower surface of the handling socket extends 8.35 in. below the bottom of the cover. Twelve holes are drilled in the cover and are sized to accommodate the closure bolts. The CCC atmosphere is free to pass between the storage tubes, but a metal Helicoflex seal is provided between the container and the cover to establish container confinement. An impact limiter is located on the bottom of the handling socket closure cup. Its function is to reduce the loading to the cover that could occur during the CCC drop accident into the ISC due to the fuel assembly in the center storage location rebounding and hitting the cover. The top surface of the cover is provided with rigging attachment points for empty container handling, a test port for seal leak-rate verification, and a sample port for container atmosphere sampling.

The CCC cover design provides a closure with a cover that mates to the CCC body, crushing a metallic O-ring between them, which provides a primary boundary for storage of the spent fuel. This boundary is not credited for confinement. Leak-testing requirements assure that the CCC will function as an effective "canning" barrier. Each CCC body and closure will be certified at the manufacturer to a leak rate of $\leq 1 \times 10^{-3}$ scc/sec. After the CCC is loaded with spent fuel in the interim examination and maintenance (IEM) cell, a pressure decay test to

$\leq 1 \times 10^{-1}$ scc/sec is performed on the seal to verify correct installation of the metal Helicoflex seal. Additionally, each fuel storage tube is closed at the bottom with a nickel alloy cup to prevent potential caustic fission product solutions from degrading the ISC confinement liner in the unlikely event of a leak out of a fuel pin. Additional features of the CCC that provide "canning" assurance are as follows: (1) all pressure boundary welds are required to be full penetration and penetrant inspected during the fabrication process; and (2) each CCC is hydrostatically tested to the design pressure of 105 lb/in^2 gauge, which is the pressure resulting from 100% fission gas release of seven fuel assemblies.

D4.3.1.3 Core Component Container Functional Requirements.

Handling/Drop – The CCC is analyzed as an integral part of the FFTF cask/container system. The CCC shall withstand the induced loads from all normal operation and handling/drop accidents. The CCC structural integrity required to maintain criticality geometry control shall not be compromised.

Cask Tipover – The CCC is analyzed as an integral part of the FFTF cask/container system. This analysis is discussed in Section D4.4.1. The CCC structural integrity required to maintain criticality geometry control shall not be compromised.

Fuel Rod Rupture – The CCC is analyzed as an integral part of the FFTF cask/container system. The CCC shall withstand the induced loads from rupture of all pins in seven fuel assemblies. This analysis is discussed in Section D4.4.1. The CCC structural integrity required to maintain criticality geometry control shall not be compromised.

Seismic –The CCC is analyzed as an integral part of the FFTF cask/container system. The CCC shall withstand the 200 Area design basis earthquake (DBE) of 0.26 g, as discussed in Section D4.4.1. The CCC structural integrity required to maintain criticality geometry control shall not be compromised.

Fire – The CCC is analyzed as an integral part of the FFTF cask/container system. The atmosphere internal to the ISC can be either dry argon or dry helium gas. For the different inerting cases evaluated, the maximum inner-cavity wall (liner) temperature limit is 495 °F (argon) or 640 °F (helium), respectively, at the fuel mid-plane (78.5 in. from the top of the CCC) to satisfy a maximum pin cladding temperature limit of 900 °F. The CCC structural integrity required to maintain criticality geometry control shall not be compromised.

D4.3.1.4 Core Component Container System Evaluation.

The CCC is designed to "can" the spent fuel. This feature will allow for effective retrieval of the spent fuel in the event of excessive cladding degradation, but does not require a leaktight confinement boundary. This feature was provided for personnel protection during retrieval of the CCC from the ISC after storage. The CCC is not designated as a confinement boundary for storage or onsite shipment. Consequently, a demonstration of confinement requirements was not imposed on the CCC "canning" boundary.

10 CFR 72 requires two independent barriers be maintained between the fuel and the environs. 10 CFR 72.122(h)(1) states that the spent fuel cladding is considered the primary confinement and must be protected during storage against degradation that leads to gross

ruptures or the fuel must be otherwise confined, "canned," such that degradation of the fuel during storage does not pose operational safety problems with respect to its removal from storage.

As described in memo 18200-SG-032, *Core Component Container White Paper* (Guttenberg 1995), even though the FFTF spent fuel cladding will meet the requirements noted above upon initial placement into the dry storage cask or ISC, the FFTF spent fuel storage system also "cans" the fuel within the CCC. The function of the CCC is to provide added assurance that the long-term primary boundary of the FFTF spent fuel storage system can maintain performance equivalent to commercial spent fuel cladding. The approach to "can" the fuel within the CCC was implemented due to: (1) the concern that cladding degradation by a phenomenon known as "hot cell rot," which results in fuel rubblization, may occur during the dry storage life, and (2) a lack of long-term data specific to a dry cask storage environment for the FFTF spent fuel.

The concern with "hot cell rot" is based on several isolated cases of fuel cladding degradation that were observed during post-irradiation storage examination of liquid metal reactor assemblies at several U.S. facilities, as well as in Europe and Japan. In all observed cases, the phenomenon was strictly associated with degradation of the fuel cladding. The phenomenon by itself does not produce gross degradation of the oxide fuel. Most documented instances were associated with storage or handling in air, or where air and moisture in-leakage was not strictly controlled. "Hot cell rot" was never observed during IEM cell fuel handling at FFTF.

The cladding degradation mechanism associated with "hot cell rot" is thought to be a form of caustic stress-corrosion cracking. The irradiated fuel cladding can have large residual stresses during its storage period, and the presence of minute amounts of residual sodium hydroxide on pin surfaces and crevices after cleaning with a moist gas-water rinse process is possible. Exposure to moisture and air could create conditions favorable to stress-corrosion cracking. The observed cladding degradation associated with "hot cell rot" ranges from scattered cracks along the fuel pin to cracking around the entire circumference of the fuel pin in the most severe case. This cracking can result in structural failure during handling of individual fuel pins; however, the experience to date has been that the cladding provided fuel retention even for the most severe case.

These isolated failures are not expected to be indicative of how the FFTF fuel will perform in "leaktight" dry storage. This is because the leaktight ISC limits the in-leakage of oxygen and moisture impurities to extremely small amounts. Preventing in-leakage of moisture and air effectively limits the conditions required for stress-corrosion cracking of the cladding to occur. Extremely small amounts of impurities are contained in the relatively pure inert backfill gas. These impurities are trapped inside the storage cask when it is sealed and can only react with a finite number of molecules before the oxidation/degradation reaction stops. Because the cask is leaktight, it prevents further significant introduction of impurities.

Even though limiting air and moisture in-leakage will effectively limit "hot cell rot" during dry storage, there remain inherent differences between the FFTF spent fuel and commercial spent fuel that prevent direct correlation of acceptable dry storage conditions. These differences include higher burn-ups of FFTF fuel that create different pin pressure characteristics and more concentrated fission product inventories, different irradiation histories and conditions, different cladding materials, and the use of sodium versus water coolant. Additionally, there are no long-term data for storage of FFTF fuel in dry cask storage conditions.

The CCC canning criterion is based on comparison to the expected storage performance of commercial fuel cladding. Additionally, according to WHC-SD-FF-ES-024, *Engineering Studies Performed for the Interim Storage Cask*, studies on commercial fuel cladding performance in long-term dry storage indicate that some failure of fuel rods can be expected in dry storage and that the resultant defects are in the form of pinholes or hairline cracks that vary in size from 1 to 30 μ. A CCC leak rate of 1×10^{-3} scc/sec correlates to a single leakage hole size of 2.5 μ in diameter using American National Standards Institute (ANSI) standard ANSI N14.5, *American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment*, Table B2. This leakage hole size determination is conservative because it assumes that only one leakage path exists (i.e., maximum hole size), and an effective "cladding replacement" boundary is easily demonstrated when compared to the commercially accepted range of 1 to 30 μ for pinhole breaches. Furthermore, the single hole leakage path assumption is extremely conservative and, in reality, the 2.5-μ hole size would be overestimated. In the case of the actual CCC design, any leakage past the seal would be through multiple minor machining marks or imperfections that run perpendicular to the seal surface. Any realistic leakage hole size would be a function of the number of surface imperfections and would be correspondingly smaller.

Further conservatism is demonstrated for particulate retention. The CCC provides a bolted closure seal at the top, and the rest of the assembly is full-penetration welded. The sealed CCC is then leak tested to 1×10^{-3} scc/sec. Therefore, it is reasonable to assume that only the seal would have minor leaks based on the unlikely 2.5-μ single leak path hole size discussed above. For a fuel fragment to approach the CCC seal, it would have to overcome gravity and traverse a highly tortuous path up and around the pin plenum region approximately 5 ft in length to the leak path. Additionally, the CCC materials have not experienced the severe environment of a reactor and would not be expected to have degraded properties. The nickel alloy cup closure located at the bottom of each CCC storage tube will retain any potentially damaging caustic mixture and prevent it from contacting the ISC confinement structure in the event of caustic material release outside a fuel pin. The CCC will provide structurally sound retrievability for a case where spent fuel degradation exceeds expectations.

Handling/Drop – The CCC was analyzed to confirm that it can safely withstand the following normal, off-normal, and accident conditions. Fully loaded CCCs (six or seven fuel assemblies, or six pin containers) are addressed in WHC-SD-FF-DA-077, *Stress and Structural Analysis of the Core Component Container*, and SNF-4790, *200 Area Interim Storage Design Basis Accident Analysis Documentation for FFTF Fuel Storage.* Partially loaded CCCs are not analyzed, and must be evaluated on a case-by-case basis.

Normal loading conditions:

- Vertical lift and set down by a crane.

Accident conditions:

- 18-ft drop of the CCC into the ISC
- 4-ft sideways drop of the ISC with the CCC inside
- 8-ft drop of the ISC with the CCC inside onto a concrete storage pad.

These conditions are analyzed for the DBA in Chapter D3.0, with the analysis summarized in Section D3.4.2.1. Material properties were evaluated using a peak CCC temperature condition of 600 °F. Normal CCC conditions were evaluated in accordance with the stress limits of the ASME Code, Section VIII, Division 2 (ASME 1989), and the accident conditions were evaluated in accordance with the stress limits of the ASME Code, Section III, Appendix F (ASME 1989).

Nuclear safety equivalency to NRC requirements would require the CCC to be designed and fabricated to ASME Code, Section III, Subsection NB or NC, for providing criticality geometry control. This was identified during final reviews for initial loading of the FFTF fuel in 1996. A decision was made, with DOE approval, to continue with the ASME Section VIII container since it provided sufficient structural capacity for storage purposes. It was acknowledged that the FFTF fuel would require repackaging into an approved ASME Section III container prior to offsite transport for final disposal at a repository. This repackaging was assumed to occur in a future "hot cell" to be co-located near the 200 Area ISA and Canister Storage Building facilities.

A thermal stress analysis was performed on the CCC. The results indicate that the average temperature around the circumference of the storage tube at the mid-plane of the core component fuel section is well below the 600 °F design temperature used to bound the structural calculations. The thermal analysis, WHC-SD-FF-ER-100, *Thermal Analysis of the Core Component Container Within the Interim Storage Cask*, shows that both normal and accident conditions will result in ISC liner temperatures that are well below the limiting design requirement parameter of 495 °F. The CCC structural integrity required to maintain criticality geometry control was not compromised.

Cask Tipover – The CCC is an integral part of the FFTF cask/container system and is analyzed for the DBA in Chapter D3.0. This analysis is summarized in Section D3.4.2.3 and discussed in Section D4.4.1. Analyses performed for the ISC tipover show that induced stresses and loads are bounded by the 4-ft drop onto an unyielding surface analyzed in Section D3.4.2.1. The CCC structural integrity required to maintain criticality geometry control is not compromised.

Fuel Rod Rupture – The accident condition of all fuel rods being ruptured is analyzed for the DBA in Chapter D3.0, and the analysis is summarized in Section D3.4.2.4. The inventory of fission gas and those gases produced during 50 years of radioactive decay are analyzed. The analysis assumes the worst-case ISC cavity average temperature of 286 °F. The maximum pressure is determined to be 62 lb/in^2 gauge for the CCC worst-case loading of seven DFAs. The design pressure of the CCC is 70 lb/in^2 gauge, thus providing a pressure margin of 8 lb/in^2 or about 11%. A finite element stress analysis was also performed for the pressure load of 70 lb/in^2 gauge (WHC-SD-FF-DA-077). The analysis determined that CCC structural integrity required to maintain criticality geometry control is not compromised. Each CCC is hydrostatically tested to the design pressure of 105 lb/in^2 gauge.

Seismic — The design basis seismic event for the CCC, as an integral part of the FFTF cask/container system, is analyzed for the DBA in Chapter D3.0 and the analysis is summarized in Section D3.4.2.5. The CCC structural integrity required to maintain criticality geometry control was not compromised, as the ISC does not slide or tip over. All seismically induced loads are well bounded by the drop impact accelerations.

Fire – The conditions of peak thermal temperature during and after the transportation DBF are analyzed for the DBA in Chapter D3.0. These analyses are summarized in Section D3.4.2.7. The fuel cladding temperature limit of 900 °F and CCC closure seal temperature limit of 500 °F are evaluated.

The accident fire results in a short-term thermal loading on the ISC. Section D3.4.2.7 summarizes the ISC temperatures in the cask during a fire. Most of the concrete remains below the 350 °F accident temperature allowable by American Concrete Institute (ACI) standard ACI 349, *Code Requirements for Nuclear Safety Related Concrete Structures*. For the conditions representing the fire accident, the large mass of concrete helps keep the interior temperatures of the cask relatively cool. The maximum liner temperature reaches 309 °F, within the allowable limit of 495 °F (with argon-inerting gas). At least 17 in. of concrete remain below the 350 °F ACI 349-90 temperature limit defined for accident conditions.

D4.3.1.5 Core Component Container Controls (Technical Safety Requirements).

The assumptions associated with the CCC that require TSRs to ensure performance of its safety function are as follows:

- Each CCC shall contain six DFAs in the outer positions, seven DFAs including the center position, or six Ident-69 containers in the outer positions. The loaded CCC can contain a mixture of fuel assemblies and pin containers such that the total is either six or seven, and the number of pin containers is five or less. No partially filled CCCs are permitted.
- There shall be no more than a total of 1,519 pins in a CCC container to protect the fuel rod rupture pressure calculation assumptions.
- Only intact fuel shall be placed into a CCC.

D4.3.2 Light Water Reactor Canister

The LWR canister structure provides required criticality geometry control and is therefore designated safety class, as identified in Table D4-2. The canister also has safety-significant functions, as discussed in Section D4.4.5. The NAC-1[3] inner LWR canister design performance requirements are established by SNF-4894, *Spent Nuclear Fuel Project Acceptance Criteria for LWR Fuel Storage System*. The canister design provides a welded closure.

[3] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

D4.3.2.1 Light Water Reactor Canister Safety Function.

The LWR canister and its internal components provide the structure that ensures criticality geometry control of the intact LWR fuel and the shell prevents water ingress, therefore precluding criticality in the storage configuration, even if damaged LWR fuel is present in the canister. The LWR canister and internal components (including support structures and the rod consolidation assembly) are designated safety class for criticality prevention only. No other safety-class functions were identified by the accident analysis (Chapter D3.0) for the NAC-1 cask system components. The analyses did identify safety-significant functions of the inner canister. The safety-significant functions for individual components of the NAC-1 cask system are discussed as an integral part of the NAC-1 cask evaluation provided in Section D4.4.5. This includes the inner canister, the NAC-1 cask, and the International Standards Organization (ISO) shipping/storage container. Specific individual component evaluation is addressed in Section D4.4.5.4.

The DBAs identified in Chapter D3.0 that could impose structural loads or accelerations on the LWR canister inside the NAC-1 cask also have the potential to change the geometry of intact or damaged fuel (Table D4-1 and Table D4-2). CSERs discussed in Chapter D6.0 credit the inner canister with providing the structure that maintains the critically safe geometry for intact fuel. In addition, the inner canister remains leak tight, preventing water intrusion such that criticality is incredible during storage, even if damaged fuel is present. The functional requirements and evaluations for the LWR canister (Sections D4.3.2.3 and D4.3.2.4) include summary discussions of the bounding DBAs using the following analyses:

- Handling/drop – Maintain confinement of radioactive materials after a credible drop (safety significant) within the NAC-1 cask, maintain structural integrity to provide criticality geometry control (safety class), and remain leaktight.

- Fuel rod rupture – Maintain confinement of radioactive materials (safety significant), maintain structural integrity to provide criticality geometry control (safety class) at the pressure generated, and remain leaktight.

- Seismic – Withstand seismic accelerations without loss of confinement (safety significant), maintain structural integrity to provide criticality geometry control (safety class), and remain leaktight.

- Tornado/wind – Withstand tornado pressure differential without loss of confinement and remain leaktight.

- Fire – Withstand transportation DBF conditions inside the NAC-1 cask without loss of confinement (safety significant) or exceeding temperature limits for fuel cladding or container components to maintain structural integrity for criticality geometry control (safety class) and remain leaktight.

The NAC-1 cask provides a passive structure that ensures the integrity of the inner canister structure will be maintained.

D4.3.2.2 Light Water Reactor Canister System Description.

The commercial LWR fuel storage system consists of six storage units, each comprised of an LWR canister, a NAC-1 cask, and an ISO shipping/storage container. The double welded inner canister will provide confinement of the fuel during storage, as required by 10 CFR 72. The NAC-1 cask provides a shielding overpack that provides weather and NPH protection.

The LWR canisters are designed and fabricated to the requirements of the ASME Code Section III, Subsection NB (ASME 1995), as required by the LWR canister acceptance criteria. Canister internals, provided for positioning the fuel assemblies, and the container for consolidation of the loose rods fall under ASME Code, Section III, Subsection NG. The canisters were designed for a life expectancy of 75 years.

The LWR canister provides the confinement boundary for the 300 Area LWR fuel. The canister is fabricated from a 12-in. stainless steel pipe with welded base cap and top closures. It has a nominal outside diameter of 12.75 in., with a maximum outside diameter of 13.375 in. at the end cap, and a total length of 177.36 in. The bottom cap is machined, so the pipe-to-cap weld is inspectable at the side of the canister. The closure lid contains a penetration that allows the canister to be evacuated, filled with helium, and leak tested to the requirements of ANSI N14.5.

The canisters provide criticality geometry control such that a loaded canister will have a k_{eff} less than 0.95 when fully moderated and reflected. Criticality is incredible if flooding within the canister is precluded, crediting the leaktight configuration of the LWR canister as an independent feature. The design weight of the canister is 1,250 lb. The maximum loaded weight of the canister is not to exceed 3,300 lb. The maximum internal design pressure of the canister is 75 lb/in^2 gauge, tested to 133% of the design pressure during fabrication in accordance with ASME Code, Section III, Division 1, Subsection NB. The final end closure weld made after the canister is loaded with spent fuel is not required to be pressure tested as identified in ASME Boiler and Pressure Vessel Code Case N-595-1.

The canister used for pressurized water reactor (PWR) assemblies uses a stainless steel inner basket tube supported by aluminum ribs to support the fuel assembly. The inner basket is an 8.49-in. square tube 160.76 in. long. The tube has a wall thickness of .120 in. and is supported by four aluminum spacer ribs between the square tube and the cylindrical LWR canister. The ribs have maximum dimensions of 7.5 in. by 1.7 in. by 163.4 in. long. These internal components of the LWR canister are also designated safety class for geometry control.

The canister used for loose boiling water reactor (BWR) and PWR rods incorporates a rod consolidation container to contain the rods (Figure D2-14) and a basket assembly for support. The consolidation container is an 8-in. by 8-in. box that is 168 in. long. The lid of the box is hinged and secured closed with 24 screws. The box is closed on the ends with perforated plates to allow helium flow during canister evacuation and backfill operations. These internal components of the LWR canister are also designated safety class for geometry control. The NAC-1 cask provides shielding and protection of the inner canister. The structures of the NAC-1 cask are constructed of stainless steel. The cask cavity is formed by the inner shell, which is a 14.125-in. outside diameter, 0.3125-in. thick stainless steel shell. The upper end of the shell is welded to the cask cavity flange; the bottom end of the shell is welded to the cask bottom casting. Surrounding the inner shell of the cask is a nominal 6.6875-in. thick annulus of chemical-grade lead (gamma) shielding. The lead is shaped such that approximately 5 in. from the bottom and 30 in. from the top, the thickness is reduced to 5.4375 in. There is an annular void, 5 in. long by 1.25 in. thick, at the bottom end of the gamma shield to allow for any lead expansion during the fire accident. The upper axial shaping is accomplished by reducing the diameter of the outer shell 2.50 in. over a 30-in. length. The lead/steel interface of the inner and outer shell have axial copper fins that are imbedded in the lead and welded to the inner and outer shells to transfer heat across the interface with a minimum temperature gradient (see Figure D2-10). Additional NAC-1 cask design and fabrication details are provided in Section D4.4.5.2.

The 300 Area LWR irradiated fuel inventory addressed in this document consists of five PWR assemblies and consolidated BWR and PWR individual rods. Each PWR assembly is packaged in an individual cask in the assembly configuration. The individual rods are packaged in a single cask in a consolidation container that fits within the inner canister. Characterization of the fuel inventory is discussed in Section D2.5.1.3.

D4.3.2.3 Light Water Reactor Canister Functional Requirements.

Handling/Drop – The inner canister and its internal components are analyzed as an integral part of the NAC-1 cask/container system. The NAC-1 cask/container system, including the canister, shall withstand the induced loads from all normal operation, handling, and storage drop accidents sufficient to ensure that the inner canister structural integrity required to maintain criticality geometry control shall not be compromised, and the canister shall remain leaktight to preclude water intrusion.

Fuel Rod Rupture – The canister shall withstand the internal pressure resulting from rupture of all contained fuel rods without compromising the canister structural integrity required to maintain criticality geometry control, and the canister shall remain leaktight to preclude water intrusion.

Seismic – The inner canister and its internal components are analyzed as an integral part of the NAC-1 cask/container system. The NAC-1 cask/container system, including the canister, shall withstand any induced stresses resulting from the design basis seismic event, with sufficient integrity to ensure that the inner canister structural integrity required to maintain criticality geometry control is not compromised, and the canister shall remain leaktight to preclude water intrusion.

Tornado/wind – Withstand tornado pressure differential without structural damage to the LWR canister, which could result in loss of criticality geometry control, and the canister shall remain leaktight to preclude water intrusion.

Fire – The inner canister and its internal components are analyzed as an integral part of the NAC-1 cask/container system. The NAC-1 cask/container system, including the inner canister, shall withstand temperatures resulting from the DBF without compromising the canister structural integrity required to maintain criticality geometry control. This includes a requirement to not exceed the maximum allowable fuel cladding temperature limit to assure that the fuel remains in the analyzed geometry, and the canister shall remain leaktight to preclude water intrusion.

D4.3.2.4 Light Water Reactor Canister Evaluation.

The LWR canister and its internal components provide the structure that ensures critical geometry control of the LWR fuel during storage at the 200 Area ISA and are designated safety class for criticality safety reasons only. The criterion for criticality safety is that a k_{eff} less than 0.95 be demonstrated for all conditions, incorporating any bias/uncertainties associated with the analysis.

Three CSERs were prepared for the interim storage of the NAC-1 casks in the 400 Area ISA. Chapter D6.0 addresses specific criticality aspects of 200 Area LWR fuel storage at the ISA as individual casks and in the storage array. The LWR canister is required for criticality geometry control for intact fuel and must remain leaktight to preclude water intrusion for storage of damaged fuel, and is therefore designated safety class. Structural integrity of the PWR fuel assembly was originally assumed by the criticality analysis to retain the fuel material within the square cross-sectional geometry of the fuel assembly and basket contained in the inner canister, or the inner canister must remain leaktight to prevent water intrusion such that geometry control is not required within the canister. Specific structural analysis of PWR assemblies need not be performed since this has been addressed for commercial fuels in a Lawrence Livermore National Laboratory document UCID-21246, *Dynamic Impact Effects on Spent Fuel Assemblies*. The document was prepared to assess the effects of dynamic impacts (to be expected from cask drop or similar incidents) on the integrity of fuel rod cladding for zircaloy-clad LWR spent fuel assemblies during cask handling and storage. The study concluded that commercial LWR fuel assemblies will not yield or buckle at the impact *g* loads calculated for the NAC-1 cask, and their structural integrity can be assumed for safety purposes. The canisters for the PWR assemblies, however, provide defense in depth for criticality safety due to the provision of safety-class support for retention of the fuel assembly geometry and the prevention of water intrusion, which also precludes criticality within a non internally flooded canister.

The consolidated individual BWR and PWR fuel rods that are stored in a NAC-1 cask are packaged in a canister with internal components that retain the cross-sectional geometry limits required for criticality control. The loose rods are packaged in an 8-in. by 8-in. box that is 168 in. long. The lid of the box is hinged and secured closed with 24 screws. The box is closed on the ends with perforated plates. The consolidation box provides the limiting dimensions for criticality geometry control. The individual rods remain oriented in a close packed restricted configuration similar to the rods in the fuel assemblies and are bounded by the assembly buckling loads and conclusions. The consolidation box can adequately contain damaged fuel if it is present. The leaktight canister also prevents water intrusion under all DBA scenarios.

Handling/Drop – The LWR canister and its internal components are analyzed as an integral part of the cask/container system in which the NAC-1 cask provides a passive barrier to stresses resulting from storage/handling drops. Evaluation of the cask/container system handling/storage drop-induced stresses is provided in Section D4.4.5. NAC-E-804, *Safety Analysis Report for the NFS-4/NAC-1, Spent-Fuel Shipping Cask*, provides evaluations that examine hypothetical accidents that conservatively bound the 200 Area ISA accident-induced stresses. The details of the analysis are provided in Chapter D3.0 and summarized in

Section D3.4.2.1. As required by Title 10, *Code of Federal Regulations*, Part 71, "Packaging and Transportation of Radioactive Material" (10 CFR 71), calculations were documented for the types of analyses discussed below. The original NAC-1 SAR-referenced sections of 10 CFR 71 (1971 edition) are no longer directly citable; however, the requirements are the same as those listed in the following statements.

Cask analyses evaluated to hypothetical accident conditions (10 CFR 71.73) include the following:

- Analyze a free drop of the cask through a distance of 30 ft onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected;
- Followed by a free drop of the cask through a distance of 40 in., in a position for which maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar mounted on an essentially unyielding, horizontal surface. The bar must be: (1) 6 in. in diameter, with the top horizontal and its edge rounded to a radius of not more than 0.25 in., and (2) of a length as to cause maximum damage to the cask, but not less than 8 in. long. The long axis of the bar must be vertical.

The design of the NAC-1 cask system is predicated on the ability of the cask to transport 150-day cooled PWR spent fuel. The thermal energy of the design basis PWR fuel imposed temperature and pressure conditions on the cask structure that produced stresses that were significantly greater than the stresses imposed by the thermal energy and pressure of the 300 Area LWR fuel rods in the inner container. The calculations provided for the NAC-1 transportation calculations are therefore conservative for 200 Area ISA storage conditions.

Evaluation of the 400 Area ISA storage configuration included results from the NAC-1 SAR (NAC-E-804) and additional analyses specific to the ISA. The evaluation of the analyses summarized in Section D.3.4.2.1 demonstrated the following:

- Conclusion for the 30-ft free fall analyses – The NAC-1 SAR analyses concluded that the integrity of the NAC-1 cask would be maintained, although significant damage would occur to the exterior, sacrificial structures of the NAC-1 cask.
- Conclusion for the 40-in. free fall onto a mild steel pin analyses – The NAC-1 SAR analyses concluded that the integrity of the NAC-1 cask would be maintained. Minimal damage to the cask, namely the localized deformation and/or slight puncture of the shield tank, would be incurred.

The LWR canister was designed and analyzed in accordance with the requirements of the ASME Code, Section III, Subsection NB (ASME 1995), as required by NUREG/CR-3854, *Fabrication Criteria for Shipping Containers*, for the Category I component safety group and 10 CFR 71. The canisters are designed for an operational environment inside the NAC-1 cask body under both normal and accident conditions of transport per 10 CFR 71, and normal and off-normal conditions of storage per 10 CFR 72.

The analysis of the LWR canister for the postulated accident conditions of transport investigated the integrity of the canister to a 96 *g* axial impact deceleration load and the probable initiation of buckling. Fatigue analysis was also performed to predict inner canister limitations to mechanical and thermal cyclic loading conditions. The analysis shows that the design of the LWR canister satisfies the required design checks with an acceptable margin of safety to ensure that the stresses induced will not compromise the canister structural integrity required to maintain criticality geometry control.

Fuel Rod Rupture – The fuel-rod rupture analysis is provided in Chapter D3.0, and the results are summarized in Section D3.4.2.4. The Chapter D3.0 analysis includes calculation of the maximum normal operating pressure for the LWR canister and assumes cask cavity temperature as a result of the hottest summer day (115 °F ambient), along with fuel rod rupture of all rods. The pressure increase within the cask cavity is due to both a temperature increase and 100% rod rupture. This pressure is calculated in SNF-4794, *200 Area Interim Storage Area Design Basis Accident Analysis Documentation for LWR Fuel*, which addresses the fission gas generated during irradiation, initial pressurization within the rods, and the gas within the canister cavity from loading and leak-testing activities.

The pressure within the cavity due to rod rupture at maximum normal conditions is well within the design pressure of 75 lb/in^2 gauge.

Worst-case pressures are also calculated for 100% rod rupture during the DBA fire conditions. This accident condition pressure is also within the design pressure for the inner canister, and the stresses induced will not compromise the canister structural integrity required to maintain criticality geometry control and will maintain the leaktight integrity of the canister.

Seismic – The seismic analysis is provided in Chapter D3.0 and is summarized in Section D3.4.2.5. As noted, stresses induced in the NAC-1 cask and the LWR canister during the 200 Area ISA DBE are within the cask system design capabilities. The LWR canister safety-class evaluation calculations consider all components to be part of an integral NAC-1 storage system.

Seismically induced loads are bounded by the drop impact accelerations above. The drop impact evaluation concludes that the cask system is able to withstand any induced stresses from handling/drop events without compromising the structural integrity of the LWR canister or internal components required to maintain criticality geometry control and a leaktight configuration.

Tornado – The LWR canister design pressure is much greater than the 0.9 lb/in^2 gauge pressure differential of the design basis tornado (DBT). This minimal differential pressure cannot generate structural damage to the LWR canister.

Fire – An analysis of the thermal response of the storage system to the DBF is provided in Chapter D3.0 and summarized in Section D3.4.2.7. The NAC-1 cask SAR (NAC-E-804) determined that the cask supported an internal thermal load in excess of the LWR fuel and presented analysis of the transportation fire event, which concluded that the cask integrity is preserved. That analysis provides a bounding case for use of the NAC-1 cask for storage of 200 Area LWR fuel. The Chapter D3.0 analysis concludes that the stresses induced will not compromise the canister or internal component structural integrity required to maintain criticality geometry control and a leaktight configuration.

D4.3.2.5 Light Water Reactor Canister Controls (Technical Safety Requirements).

The assumptions associated with the LWR canister that require TSRs to ensure performance of its safety function of criticality geometry control are as follows:

- The number of individual fuel rods in a LWR canister does not exceed a maximum of 179 PWR rods, or 96.5 BWR rods consolidated with 17 PWR rods.
- The individual loose rods must be retained within the rod consolidation container.

D4.4 SAFETY-SIGNIFICANT STRUCTURES, SYSTEMS AND COMPONENTS

Safety-significant items are SSCs, including portions of storage systems, whose failure could adversely affect the safety and health of the collocated worker. Detailed definitions of safety-significant SSCs are provided in Section D4.1.

The SSCs credited with a safety-significant (or NRC Important-to-Safety Category B) function in Chapter D3.0 are described in the subsections that follow.

D4.4.1 Fast Flux Test Facility Interim Storage Cask System

This section discusses the safety-significant aspects of the ISC and CCC. The criticality safety aspects of the CCC, which relate to safety-class considerations, are discussed in Section D4.3.1.

D4.4.1.1 Safety Function.

The ISC is designed to provide secondary confinement for the fuel and structural protection for the CCC. It also provides passive heat removal and radiological shielding protection for site personnel by limiting the dose rate at normally accessible surfaces to acceptable levels (not credited safety functions). As shown in Table D4-2, the ISC is credited with safety-significant functions for the following DBAs identified in Chapter D3.0:

- Handling/drop – Maintain confinement of radioactive materials after a credible drop and provide passive protection of the CCC such that it retains structural integrity.
- Mobile crane fall – Maintain structural integrity sufficient to maintain confinement of radioactive materials after a crane fall.
- Cask tipover – Maintain confinement of radioactive materials after the crane fall and provide passive protection of the CCC such that it retains structural integrity.
- Fuel rod rupture – Maintain confinement of radioactive materials after the rupture of all fuel pins.
- Seismic – Withstand seismic accelerations without loss of structural integrity sufficient to maintain confinement, and without tipover or sliding.
- Tornado/wind – Withstand DBT winds (excluding DBT missiles) without sliding or tipover. Also withstand the design basis wind and wind-driven missiles established

for the 200 Area ISA without loss of structural integrity sufficient to cause loss of confinement.

- Fire – Withstand transportation DBF conditions without loss of structural integrity sufficient to cause loss of confinement or exceeding temperature limits for fuel cladding or cask components.

In addition, the ISC provides a confinement boundary for all normal conditions and abnormal events.

As shown in Table D4-2, the CCC is credited with safety-class criticality control functions for the DBAs identified in Chapter D3.0 and previously discussed in Section D4.3.1, as follows:

- Handling/drop
- Cask tipover
- Fuel rod rupture
- Seismic
- Fire.

D4.4.1.2 System Description.

The ISC is an aboveground concrete and steel shielded, top-loading spent fuel storage cask that will be used to provide safe interim dry storage of a CCC with FFTF spent fuel assemblies or pin containers for a period of up to 40 years at the 200 Area ISA. One CCC can be stored in the cavity of each ISC. The ISC has been designed and fabricated to meet the requirements of WHC-S-4110, *Specification for FFTF Interim Storage Cask*, in accordance with 10 CFR 72. "Canning" of the spent fuel is provided by the CCC, as discussed in Section D4.3.1. The ISC is designed to provide secondary confinement for the fuel and environmental protection for the CCC. The ISC also provides passive heat removal and radiological shielding protection for site personnel by limiting the dose rate at normally accessible surfaces to acceptable levels. A weather protection cover is installed on each ISC in the ISA.

The ISC design consists of an all-stainless steel internal confinement structure surrounded by steel and concrete shielding. The fully loaded cask, including a loaded CCC with contents weighing up to 5,000 lb, the closure hardware and the weather cover, weighs a maximum of 114,200 lb. Outer cask dimensions are 85 in. in diameter by 181 in. tall, excluding the weather cover. The internal cavity of the ISC is 21 in. in diameter and 147 in. tall, and will accept one CCC. This cavity, which is formed by a 1.5-in. thick stainless steel cylinder and 8-in. thick bolted top and welded bottom closure plates, provides the confinement boundary. An additional cover plate may be seal welded over the bolted closure after receipt at the 200 Area ISA to enhance the long-term storage configuration. This cover plate would be seal welded to the vertical cylindrical stainless steel liner around and above the bolted shield closure plug. The lower end of this existing liner is welded to the top confinement flange of the ISC cavity, and there is a 3-in. recessed space above the bolted closure below the existing weather cover. Although it is not anticipated that this cover plate would be removed during the storage period, the seal weld located in this space is accessible with the weather cover removed and could be readily ground out to restore the ISC to its original configuration for future repackaging or for access to test ports in the closure plug.

The ISC confinement boundary design and analysis were performed by General Atomics, which holds an ASME Certificate of Authorization N for design and overall fabrication responsibility for ASME Code, Section III, Division 1 and 2 components. The ISC design analysis and material properties were based on the requirements of the ASME Code, Section III, Subsection NC (ASME 1989). Fabrication of the ISC is based on the ASME Code, Section III, Article NC-4000. The ISC confinement boundary is constructed of ASTM materials but was not required to be stamped or built by a certificate holder. Examples of using the ASME Code, but not requiring a code stamp, can be found in NUREG/CR-3854 and NUREG/CR-3019, *Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials*. The fabricator was not required to be a certificate holder as the design was performed by a vendor with an ASME Certificate of Authorization N for design and overall fabrication responsibility for ASME Section III, Division 1 and 2 components, and the N stamp was not required. Additionally, all ISC design and fabrication activities were performed using a 10 CFR 72 quality assurance program or equivalent.

The bottom of the ISC cavity is fitted with an aluminum crush pad to limit CCC impact loads in the unlikely event that it is dropped into the ISC during loading. The surfaces of the cavity are finished to remove burrs, sharp corners, and weld beads that could potentially interfere with cask loading operations. The confinement structure described above is also surrounded by annular steel shield plates that are surrounded by concrete reinforced with rebar. As discussed in Document No. 910683, *FFTF Spent Fuel Interim Storage Cask Design Analysis Report* (General Atomics 1995), the concrete shielding structure is designed to meet ACI 349-90 requirements. The concrete design mix was selected to ensure strength and long life over the range of temperature conditions expected during normal operations, and the more extreme short-term temperatures that could occur during off-normal or accident conditions.

The ISC heat dissipation is totally passive. All required heat removal can occur by conduction, thermal radiation, and convective cooling of the outer surface. The ISC is also provided with a passive cooling system that removes heat by using an internal natural circulation airflow system. The airflow system is formed by two inlet ducts, an annular gap between the confinement boundary cylinder and the inner shield, and two outlet ducts. The inlet and outlet ducts are steel-lined penetrations through the concrete that take non-planar paths to minimize radiation streaming. These penetrations consist of two 4-in. outside diameter ducts that supply air to the bottom of a 0.75-in. wide annulus between the confinement boundary cylinder and the inner shielding cylinder. Natural convection circulates air up the annular space and out two similar ducts at the top of the annulus. This overall heat removal system is designed to limit localized concrete temperatures to below 200 °F on the warmest normal condition day, with the bulk of the concrete maintained below 150 °F. The ISC thermal analysis for the extreme ashfall accident (General Atomics 1995) shows that even if the ducts become plugged and natural convection cooling is eliminated, these limits will still be met. Therefore, operability of the cask ventilation system is not required to be monitored to ensure that cask thermal limits are not exceeded.

Based on the white-painted exterior surface of the ISC, the solar absorptivity of the cask is taken to be 0.3 and the emissivity is 0.9. The emissivity value is based on heavily oxidized paint. The only possible effect that could result from degradation of the paint is a possible increase in the absorptivity, but this case is bounded by the analyzed ashfall accident case

(General Atomics 1995), which demonstrates normal limits are not exceeded. Therefore, the ISC paint color is not a safety requirement. The off-normal severe cold is used to evaluate the thermal distributions of the ISC for the -27 °F lowest ambient temperature reported for the Hanford Site. All temperatures are within limits, and the thermal gradients are less than the normal condition case with no wind.

The ISC provides several layers of shielding to ensure worker as-low-as-reasonably achievable (ALARA) radiation protection. The first layer of shielding consists of a 3.0-in. thick carbon steel clamshell around the entire length of the cavity. Another partial length, 4.0-in. thick, carbon steel clamshell shield is provided for additional shielding at the cavity mid-plane where the fuel section is located. The clamshell shields are designed with studs that attach to the reinforced concrete cylinder shield. This shield is a minimum of 21.25 in. thick for additional radial shielding. Supplemental axial shielding is provided by 4.0-in. thick plates below the bottom head and below the upper closure. A design dose rate of 2 mrem/h at accessible surfaces for normal conditions and 200 mrem/h at the (inaccessible) bottom head is defined in WHC-S-4110. Shielding acceptance criteria in the design specification allowed a localized dose rate of 5.0 mrem/h to account for potential shielding imperfections and localized hot spots.

The design description of the CCC is presented in detail in Section D4.3.1.

D4.4.1.3 Fast Flux Test Facility Interim Storage Cask System Functional Requirements.

The ISC containing a CCC shall be able to withstand the accident loads associated with handling and drops, mobile crane falls, tipovers, fuel rod ruptures, seismic events, tornado/wind events, and fire/thermal events. In addition to those loads, additional handling accident loads were evaluated to fully envelop all ISC load conditions. The ISC shall withstand these design-basis accident loads to the extent that the reduction in shielding is not sufficient to increase the external dose rate to more than 1,000 mrem/h at 1 m from the external surface of the package specified in WHC-SP-4110, and the ISC confinement integrity is maintained (i.e., stress levels in the ISC confinement boundary do not exceed the levels specified for Level D service limits of the ASME Code, Section III [ASME 1989]). Each type of accident load is discussed below.

Confinement – Two confinement boundaries shall be maintained for the fuel in normal and accident conditions. The accident loads and conditions under which confinement must be maintained are identified in Section D4.4.1.1. The ISC shall be one of the confinement boundaries, and the intact fuel cladding shall be the other confinement boundary. The cask must be designed to provide redundant sealing of confinement systems.

Handling/Drop – The FFTF cask system shall be analyzed to remain functional with no loss of confinement or structural integrity for the following handling and drop scenarios:

- A free drop of 40 in. striking the top end of a vertical, cylindrical, mild steel bar mounted on an unyielding horizontal surface in the worst-case orientation.
- A free drop of 4 ft onto a flat, unyielding, concrete horizontal surface, striking the surface in a position for which maximum damage is expected.
- A free drop of 8 ft onto a flat, 1.5-ft thick reinforced concrete horizontal surface, striking the surface in a position for which maximum damage is expected.

- A drop of a fully loaded CCC from a height of 18 ft into an ISC.

Loads shall be combined such that normal loads from pressure, temperature, and dead weight act in combination with all other loads. No two off-normal or accident events are postulated to occur simultaneously.

Mobile Crane Fall – The ISC shall be able to withstand direct impact of the mobile crane boom without loss of structural integrity sufficient to maintain confinement if a crane failure occurs.

Cask Tipover – The ISC shall be able to withstand the induced load from a tipover accident without loss of confinement, loss of shielding beyond accident limits, or loss of criticality configuration geometry control by the CCC.

Fuel Rod Rupture – The ISC design pressure of 36.7 lb/in^2 gauge shall not be exceeded by fuel rod rupture regardless of thermal conditions.

Seismic – The ISC containing the CCC shall be able to withstand the 200 Area ISA DBE without loss of structural integrity sufficient to maintain confinement and without sliding or tipover.

Tornado/Wind – The ISC shall be able to withstand the DBT winds (excluding missiles) without loss of structural integrity sufficient to maintain confinement and without sliding or tipover. The ISC shall also be able to withstand the design basis wind and wind-driven missiles established for the 200 Area ISA.

Fire – The design basis storage fire is bounded by the design basis transportation fire of 10CFR71.73(c)(3) per SNF-4932, *Fire Hazards Analysis for 200 Area Interim Storage Area*, with the following requirements:

- Fuel cladding temperatures shall not exceed 900 °F to preclude cladding breach.
- The cask shall be designed to provide adequate heat removal capacity without active cooling systems (10 CFR 72.236[f]).
- The atmosphere internal to the ISC can be either dry argon or dry helium gas. For each inerting case that is evaluated, the inner-cavity wall (liner) maximum temperature limit is 495 °F (argon) or 640 °F (helium) at the fuel mid-plane (78.5 in. from the top of the CCC) to satisfy the maximum pin cladding temperature limit of 900 °F. The acceptance criteria for the ISC peak liner temperature was based on pre-design analyses that were performed to allow the ISC design to proceed independent of a CCC thermal analysis. These criteria were established (WHC-SD-FF-ES-024) based on the 900 °F cladding limits. The ISC liner limits were determined by analysis of various inerting cases based on a worst-case concrete and steel shielded, non-ventilated, storage cask configuration containing a CCC loaded with six 250-W assemblies. For this configuration, ISC liner limits of 495 °F with argon-inerting gas, or 640 °F with helium-inerting gas, were chosen to ensure that the 900 °F cladding temperature limit would not be exceeded for any cask design. Since the CCC is inerted with argon, the 495 °F limit is conservatively imposed for the ISC liner.

- The concrete temperature limits for the ISC consist of normal localized, average, and accident temperature limits in accordance with guidelines of ACI 349-90. The ACI long-term concrete normal temperature limits are 150 °F for average temperatures and 200 °F for localized areas. The short-term accident temperature limit is 350 °F.
- The Helicoflex seal temperature limits are 700 °F for the ISC and 500 °F for the CCC. These limits are established based on the manufacturer's recommendation for the different seal materials.
- Complete blockage of the ISC ventilation ducts shall not result in temperatures that exceed any functional requirements listed above.

D4.4.1.4 Fast Flux Test Facility Interim Storage Cask Evaluation.

Confinement – The FFTF spent fuel storage system is provided with multiple systems to confine the radioactive fuel. The ISC serves as the main pressure boundary and leakage barrier to the environment to ensure that environmental, worker, and public safety is maintained and to control the storage atmosphere such that degradation of the spent fuel remains limited. The maximum permissible design leakage rate for the ISC confinement barrier is "leaktight," as defined in ANSI N14.5. At this leak rate, 1×10^{-7} scc/sec air, the stored spent fuel is not expected to degrade during long-term dry storage, as the leaktight boundary prevents oxygen and moisture from entering the cask and contaminating the inert atmosphere. As such, there is no known mechanism for accelerated corrosion of the spent fuel. The FFTF spent fuel cladding must meet the intact fuel criteria when the fuel is placed into the CCC. Because each spent fuel assembly will be washed using the sodium removal process in the IEM cell, an assessment of cladding integrity is achieved before transfer to dry storage. The sodium removal wash water will provide indication of a gross cladding failure.

There are no long-term data specific to dry storage of FFTF spent fuel in the leaktight inert dry atmosphere conditions of a dry storage cask. The ISC design is consistent with commercial spent fuel dry storage practice and restricts oxygen and moisture in-leakage to minute levels such that degradation of the spent fuel is expected to be effectively controlled over the storage lifetime. Even so, because there is no long-term data, it is assumed that the FFTF spent fuel cladding may not perform in the same manner as commercial fuel. This results in the requirement to "can" the spent fuel assemblies within the CCC.

The CCC boundary consists of a cluster of seven closed-bottom tubes that form the storage basket, with a bolted closure that contains a single metal seal. The CCC body and closure seal assembly are tested and certified at the manufacturer to provide a $< 1 \times 10^{-3}$ scc/sec leakage rate, in accordance with ANSI N14.5. Additionally, the final closure assembly seal, located at the top of the CCC approximately 5 ft above the fuel zone, is tested to a leak rate of $< 1 \times 10^{-1}$ scc/sec after loading the spent fuel in the IEM cell. The assembly test is in addition to visual verification that the seal is correctly installed. This approach ensures the fuel is effectively "canned" and exceeds the gross visible rupture criteria for cladding confinement that is allowed for commercial spent fuel.

10 CFR 72.236(e) requires the ISC to be designed for redundant sealing of the confinement systems. The ISC is designed with redundant seals in the closure lid and redundant

welds on the penetration port cover (10 CFR 72). Confinement system redundancy for the ISC is ensured by a combination of inspection techniques, which include radiographic and ultrasonic inspection, helium leak testing, and dye penetrant testing of the confinement welds. The confinement capability of the empty ISC liner assembly is assured by the manufacturer by radiographic inspection of the longitudinal and circumferential full penetration welds, and by ultrasonic inspection of the liner to upper flange and bottom plate full-penetration welds. In addition to these tests at the manufacturer, a complete helium leak test of the entire confinement liner is performed. The confinement capability of the loaded ISC is assured by helium leak-testing both closure seals after assembly and dye penetrant testing both seal welds of the penetration port cover plates.

Tests and specifications relevant to confinement integrity of the ISC at the time of cask loading include the following:

- ISC helium backfill pressure requirements
- ISC dye penetrant test of the penetration port closure welds
- ISC maximum permissible leak rate.

10 CFR 72.236(j) required the ISC to be inspected to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its confinement effectiveness. The quality standards under which the ISC was fabricated and welded provide the assurance of confinement integrity. Additionally, the ISC was pressurized and leak tested after all confinement welding was completed.

A thermal analysis states the atmosphere internal to the ISC can be either dry argon or dry helium gas. For the different inerting cases evaluated, the maximum inner-cavity wall (liner) temperature limit is 495 °F (argon) or 640 °F (helium), respectively, at the fuel mid-plane (78.5 in. from the top of the CCC) to satisfy a maximum pin cladding temperature limit of 900 °F to preclude cladding breach. The thermal analysis shows that the bounding case is reached during the ashfall case in which the peak cladding temperature reaches 800 °F (WHC-SD-FF-ER-100). The maximum seal temperature for the CCC reaches 196 °F during the ashfall condition, as indicated in the FFTF ISC design analysis report (General Atomics 1995).

The sealing system for the ISC retains its confinement capability when subjected to normal, off-normal, and accident loading conditions because there are no normal or accident conditions that will breach the structural integrity or leaktightness of the ISC.

Handling/Drop – The FFTF cask/container system is analyzed for the DBA drops in Chapter D3.0, and the analysis is summarized in Section D3.4.2.1. Several drop analyses were performed for the ISC. Under normal operations, the ISC is lifted vertically by attaching the lifting fixture to three anchor attachments imbedded in the concrete of the ISC. No side lifting is allowed. In accordance with the critical lift requirements of DOE/RL-92-36, the ISC lift points are designed to lift five times the weight of the cask without exceeding the ultimate stress of the material or three times the weight without exceeding the yield strength of the material, whichever is less. Therefore, the normal ISC lifting and handling loads will generate relatively low stresses in the lift points. The weakest link in the ISC attachment design is the anchor lug, which screws into the anchor bolt of the cask. If a lifting accident occurs whereby a lift point fails, this anchor fails before the ISC concrete embedment. In this manner, the ISC integrity or shielding is not affected.

As discussed in Section D3.4.2.1, the ISC system is shown to survive the following drop accidents:

- 18-ft drop of the CCC into the ISC
- 4-ft sideways drop of the CCC while inside the ISC
- 4-ft ISC drop onto an unyielding surface
- 8-ft ISC drop onto a 1.5-ft thick concrete pad
- 40-in. ISC drop onto a 6-in. pin.

The CCC will be contained within the ISC for the entire duration of this project, so an independent drop of the CCC is not possible. However, the CCC drop analyses are identified to demonstrate that it will survive the drops analyzed for the ISC.

The conclusion of the accident analysis is that the ISC system survives all of the analyzed drops without loss of confinement. The 4-ft ISC drop onto an unyielding surface is the bounding case. The 8-ft ISC drop onto a 1.5-ft thick concrete pad is bounded because the pad yields and absorbs much of the impact energy. Since the 200 Area ISA concrete pad is 1.5-ft thick, an 8-ft drop is tolerable without loss of confinement.

As indicated in the accident analysis in Section D3.4.2.1, TSR controls have been applied to ensure that the ISC system is not taken outside of the analyzed bounds.

Mobile Crane Fall – The FFTF cask/container system is analyzed for the DBA in Chapter D3.0, and the analysis is provided in SNF-4790 and summarized in Section D3.4.2.2. The structure of the ISC withstands the induced loads from all mobile crane fall accidents without loss of confinement function. The CCC structural integrity is not compromised.

Cask Tipover – The FFTF cask/container system is analyzed for the DBA in Chapter D3.0, and the analysis is summarized in Section D3.4.2.3. Analyses performed for the ISC tipover show that induced stresses and loads are bounded by the 4-ft drop onto an unyielding surface analyzed in Section D3.4.2.1. The CCC structural integrity is not compromised.

Fuel Rod Rupture – The accident condition of all fuel rods being ruptured is analyzed for the DBA in Chapter D3.0, and the analysis is summarized in Section D3.4.2.4. The inventory of fission gas and gases produced during 50 years of radioactive decay is expected to be 5.4 gram-atoms (moles)/DFA or 37.8 gram-atoms/ISC. This assumes that all free gases contained within the cladding are released. The analysis assumes the worst-case ISC cavity average temperature of 286 °F. The maximum pressure is determined to be 62 lb/in^2 gauge maximum for the CCC worst-case loading of seven DFAs (WHC-SD-FF-ER-100). The design pressure of the CCC is 70 lb/in^2 gauge. A finite element stress analysis is also performed for the pressure load of 70 lb/in^2 gauge (WHC-SD-FF-DA-077). The CCC structural integrity is not compromised. Each CCC was hydrostatically tested to the design pressure of 105 lb/in^2 gauge.

The ISC cavity internal pressure is also calculated for the same conditions (General Atomics 1995). The maximum ISC cavity pressure, based on the maximum cavity average temperature of 286 °F for the ashfall accident case, is determined to be 36.7 lb/in^2 gauge, which is the design pressure for the ISC.

Seismic – The design basis seismic event for the FFTF cask/container system is analyzed for the DBA in Chapter D3.0, and the analysis is summarized in Section D3.4.2.5. The CCC structural integrity is not compromised, as the ISC does not slide or tip over. All seismically induced loads are considered bounded by the drop impact accelerations.

The ISC is designed to withstand the seismic motions from the DBE conditions (0.25 *g*) for the 400 Area ISA. The FFTF ISC design analysis report (General Atomics 1995) states that the cask is a very stiff structure and consequently acts as a rigid body during an earthquake. During the maximum anticipated seismic event, the cask will not slide or tip over. Stresses generated by the seismic event are within the allowable defined limit for accident conditions. Supplemental analysis for the 200 Area ISA indicates the ISC will not slide or tip over when the 200 Area ISA seismic inputs of 0.26 *g* (identified in Table D4-2) are applied. This analysis is documented in HNF-2183, *Overturning and Sliding Assessment for the Interim Storage Cask at the 200 East Area Interim Storage Area.*

Tornado/Wind – The DBT event for the FFTF cask/container system is analyzed for the DBA in Chapter D3.0, and the analysis is summarized in Section D3.4.2.6. Wind velocities are bounded by the tornado winds, but the wind missile is also analyzed. A 15-lb., 2x4 traveling at 50 mph only penetrates the concrete surface about 1 in. The tornado winds do not result in sliding or tipover of the cask. There are no induced loads to the CCC as a result of the tornado or wind events.

Fire – The conditions of the transportation DBF and peak thermal conditions are analyzed for the DBAs in Chapter D3.0, and the analyses are summarized in Section D3.4.2.7. The fuel cladding temperature limit of 900 °F, the CCC closure seal temperature limit of 500 °F, and the ISC closure seal temperature limit of 700 °F are evaluated.

The fire accident would result in a short-term thermal loading on the ISC. Section D3.4.2.7 summarizes the ISC temperatures in the cask during a fire. Most of the concrete remains below the 350 °F ACI 349-90 accident allowable. For the conditions representing the fire accident, the large mass of concrete helps keep the interior temperatures of the cask relatively low. The maximum liner temperature reaches 309 °F, well within the allowable limit of 495 °F (with argon-inerting gas). At least 17 in. of concrete remain below the 350 °F ACI 349-90 temperature limit defined for accident conditions.

The maximum ISC closure seal temperature for the fire accident scenario reaches 200 °F, well below the allowable limit of 700 °F for the Helicoflex metal seal.

The closure bolt stress generated during the fire accident is 65,032 lb/in^2, which is much lower than the 84,000 lb/in^2 allowable. Because the bolts do not yield, sealing performance is unaffected by the fire. Therefore, the cask will remain within design limits during a fire.

D4.4.1.5 Fast Flux Test Facility Interim Storage Cask Controls (Technical Safety Requirements).

The assumptions associated with the ISC that require TSRs to ensure performance of its safety function are as follows:

- The ISC shall not be lifted more than 8 ft above the ground or storage pad.
- Each ISC shall be placed in a storage array with spacing of at least 24 in. x 44 in. between ISCs measured edge to edge. This specification applies to all ISCs to provide adequate access to the casks and to meet thermal analysis boundary conditions. A method shall be established for indicating proper placement, and proper positioning shall be verified.
- Crane loads other than ISC rigging shall not be operated over a loaded ISC.
- The ISC system may not be lifted over objects taller than 4 ft or containing radioactive materials.
- The ISC shall be handled with an approved crane. Only the Manitowoc 4000 150T crane with Model 22 80-ft boom and the Manitowoc 4100 250T crane with Model 27 80-ft boom have been analyzed in this Final Safety Analysis Report. A program shall be in place to ensure that only approved cranes are used.
- Fire loadings are to be controlled per the fire hazards analysis.

D4.4.2 Rad-Vault System

A 250 kW TRIGA experimental research reactor was operated in the 300 Area intermittently from the late 1970s until its last power run in May 1989. The reactor was manufactured by the Gulf General Atomics Company of San Diego, California, and was used primarily for neutron radiography of FFTF fuel elements and test assemblies. The fuel from the reactor core/pool storage has been removed as part of the decommissioning of the facility. The TRIGA irradiated fuel inventory consists of 99 TRIGA fuel elements and two fuel follower control rods (FFCRs). The TRIGA fuel and the fuel followers will be stored in TRIGA fuel casks and DOT-6M containers prior to shipment to the 200 Area ISA, and the sealed casks and containers will be placed in concrete Rad-Vaults[4] at the 200 Area ISA.

The concrete Rad-Vault at the ISA will contain six NRF TRIGA casks and two special DOT-6M containers containing one FFCR each. Each TRIGA cask holds up to 17 fuel elements.

D4.4.2.1 Rad-Vault Safety Function.

The Rad-Vault provides passive protection from natural phenomena events that could damage the NRF TRIGA casks or the DOT-6M containers. Based on the results of the Chapter D3.0 consequence analysis, the Rad-Vault provides no confinement function and is designated general service; however, the Rad-Vault provides structural protection of the SNF

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[4]Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.

storage containers for several accidents, and as such, is designated as NRC important-to-safety Category B. A general-service RAD-Vault is currently in service at the 400 Area ISA. The risk of use of a general-service Rad-Vault at the 200 Area ISA is considered acceptable due to the low activity SNF, as discussed in the Section D3.4.1 analysis section.

Accidents and concerns identified in the Chapter D3.0 analysis and listed in Table D4-1 and Table D4-2, as associated with the important-to-safety functions for the Rad-Vault, are as follows:

- Cask tipover – Passive design features preclude tipping.
- Seismic – Withstand seismic accelerations without tipover or sliding.
- Tornado/wind – Withstand DBT winds (excluding DBT missiles) without sliding or tipover. Also withstand design basis wind and wind-driven missiles established for the 200 Area ISA.
- Fire – Withstand transportation DBF conditions such that TRIGA casks and DOT-6M/2R containers inside do not lose confinement or exceed temperature limits for fuel cladding or container components.

D4.4.2.2 Rad-Vault System Description.

The concrete Rad-Vault, which is described in *On-Site Storage Containers and RADVAULT Storage Containers Technical Information Package* (CNSI 1992) and shown in Figure D2-3, is a low-level radioactive waste storage unit consisting of a right circular (5,000 lb/in^2) concrete cylinder with light steel reinforcement. The Rad-Vault will be placed on compacted gravel at the ISA, and top loaded with the NRF TRIGA casks and DOT-6M containers. The Rad-Vault was constructed in accordance with ACI-301, *Specifications for Structural Concrete for Buildings*, and ACI-318, *Building Code Requirements for Reinforced Concrete*. The Rad-Vault is 111 in. high, with an outer diameter of 114 in. The wall is 18-in. thick steel reinforced concrete, with an 8-in. thick bottom. The equivalent lead shielding is 3.1 in. The empty Rad-Vault weight is 63,400 lb (43,400-lb body and 20,000-lb lid), and the maximum design weight with contents is 81,400 lb (CNSI 1992). The NRF TRIGA storage system's loaded weight is 76,760 lb. The NRF TRIGA casks and DOT-6M containers and accompanying fuel weigh approximately 13,358 lb.

The Rad-Vault is a commercial item typically used for onsite storage of low-level radioactive waste at commercial nuclear power plants, but is not a licensed container. The Rad-Vault concept provides onsite storage for commercial power plants in the unlikely event that offsite disposal is limited or prohibited. An onsite engineered storage building is not required (CNSI 1992). The design life is not specified by the vendor, but the concrete vault is epoxy coated and has an estimated life, with proper maintenance, of 50 years.

The Rad-Vault is equipped with a removable lid that fully exposes the available internal storage volume. Its mating surface has a weather gasket (optional) between the main container and is sloped to minimize rain intrusion. Opposing lifting lugs are interlaced into the steel rebar and welded wire fabric and are cast into the concrete structure. Lift capacity is sufficient to allow an empty Rad-Vault to be lifted and moved by crane or to a transport trailer. The lid must be transported separately. The Rad-Vault is not intended to be lifted loaded or with the lid installed.

Sampling and/or drain capability is provided through a siphon-type arrangement that prevents leakage in and out of the Rad-Vault. Each Rad-Vault is also equipped with a pop-up vent. No hydrogen gas is expected to be vented; however, the pop-up vent and siphon will provide a vented system. The criticality evaluation documented in WHC-SD-FF-CSER-006, *Criticality Safety Evaluation Report for TRIGA Fuel Storage at 400 Area Interim Storage Area*, was performed for a flooded vault. There are no safety requirements for the vent or the drain functions. To ensure shielding uniformity and integrity, the Rad-Vault was gamma scanned. The Rad-Vault is painted internally and externally with two coats of epoxy paint.

The fabrication materials for the storage system components for the Rad Vault are as follows:

- ASTM A-615 for bar reinforcement
- ASTM A-185 for welded wire fabric
- ASTM A-36 or equivalent for embedments
- Concrete – properties (including lid): $f_c' = 5{,}000\ lb/in^2$ in 28 days, nominal density of $145 \pm 5\ lb/ft^3$, a water binder ratio of 0.39 to 0.42, and air entrainment of 5 to 7 percent. ACI-301 specifications were provided for structural concrete. Test reports were included that verified this information. (The wall and lid of the Rad-Vault passed the required gamma scan; all data is part of the Certificate of Compliance package.)

The six NRF TRIGA casks and two DOT-6M containers will be transported to the ISA in three shipments. Three different cask/container configurations in the Rad-Vault will be required to accommodate the fuel transports. The cask configuration in the Rad Vault after the first shipment of two NRF TRIGA casks is shown in Figure D2-7. After the fuel has been transported to the ISA, the impact limiter overpack used during transport will be removed and the NRF TRIGA casks will be lifted by crane into the Rad-Vault. After each placement of containers in the Rad-Vault, the lid will be placed on top by the crane.

Four NRF TRIGA casks will be arranged in the Rad-Vault, as shown in Figure D2-7, after the second shipment of two casks. The third shipment will transfer two additional NRF TRIGA casks and the two DOT-6M containers. The final Rad-Vault configuration of six NRF TRIGA casks and two DOT-6M containers is shown in Figure D2-7. The placement of the casks, containers, and the empty 55-gal drums is based on operational procedures and is not required for criticality safety. The single drum in the center after loading performs a passive function that results in a close packed array, which precludes tipover of the containers within the Rad-Vault.

A nuclear criticality safety evaluation was performed for the transportation of the NRF TRIGA casks to the ISA pad and is included in WHC-SD-TP-SARP-008, *Safety Analysis Report for Packaging NRF TRIGA Packaging*. WHC-SD-FF-CSER-006 was prepared for the interim storage of the TRIGA fuel in the ISA to address storage of NRF TRIGA casks and DOT-6M containers in the Rad-Vault. Chapter D6.0 addresses specific criticality aspects of TRIGA fuel storage at the ISA.

D4.4.2.3 Rad-Vault Functional Requirements.

Cask Tipover – The Rad-Vault shall not tip over as the result of natural phenomenon events, acting as a passive barrier and preventing damage to the TRIGA casks or the DOT-6M containers.

Seismic – The Rad-Vault shall withstand any induced stresses resulting from the design basis seismic event with sufficient integrity to act as a passive barrier and prevent damage to the TRIGA casks or the DOT-6M containers.

Tornado/Wind – The Rad-Vault shall withstand any induced stresses resulting from the DBT/wind event with sufficient integrity to act as a passive barrier and prevent damage to the TRIGA casks or the DOT-6M containers.

Fire – The Rad-Vault in the 200 Area storage location shall be capable of withstanding the DBF with sufficient integrity to act as a passive barrier and prevent damage to the TRIGA casks or the DOT-6M and 2R containers.

D4.4.2.4 Rad-Vault System Evaluation.

The Rad-Vault is the exterior container of the TRIGA fuel storage system in the 200 Area ISA storage configuration. The Rad-Vault is the only container that is exposed to a majority of the events analyzed. Resistance to the events precludes similar exposure to the TRIGA casks, the DOT-6M containers, and the NRF TRIGA fuel during storage and ensures that the appropriate inner components withstand the design basis events with no loss of confinement. The important-to-safety function is to prevent significant damage to the storage containers within the Rad-Vault.

Seismic – The analysis, summarized in Section D3.4.2.5, shows the design of the Rad-Vault is well within the DBE for the site, and seismic stability is assured. The Rad-Vault will not overturn or slide on the soil. The lid lip will not fail, and the lid will remain on the Rad-Vault. The NRF TRIGA casks inside the container will not suffer loss of integrity, and the DOT-6M containers that will be used for the FFCRs are qualified for conditions that are in excess of the DBE.

The flexure and shear stiffnesses of the Rad-Vault are also analyzed in Chapter D3.0 to determine the natural periods, and hence frequencies, of the structure under seismic excitation.

A comparison between the natural horizontal frequency of the Rad-Vault and the response spectra for the Hanford Site design basis earthquake shows that the Rad-Vault will behave like a rigid body during a seismic event, and dynamic analysis is not required.

Tornado/Wind – The Rad-Vault weighs approximately 63,400 lb. without a load and has a center of gravity located approximately at the geometric center. It is loaded with the six NRF TRIGA casks and two DOT-6M containers. The analysis, summarized in Section D3.4.2.6 and including calculations from the manufacturer (CNSI 1992), shows that an empty Rad-Vault will endure winds and pressure transients in excess of the 200 Area ISA DBT without sliding or tipover.

The Chapter D3.0 analysis indicates that individual NRF TRIGA casks or DOT-6M containers will be exposed to the risk of a tornado for a short time during the crane lift from the overpack to the Rad-Vault by the crane. However, DOE/RL-92-36, Section 3.0, "Critical Lifts," will be used during the lifts, and the critical lift procedure will be prepared to identify any weather-related limits (as required) for the load being transferred. Therefore, the time period in which the NRF TRIGA casks or DOT-6M containers might be exposed to inclement weather conditions is negligible and, therefore, is not analyzed.

The loads resulting from the design basis wind are bounded by the consequences of the DBT discussed above, with the exception of wind-driven missiles. The Chapter D3.0 analysis also determines the penetration of the Rad-Vault by a wind missile. The evaluation concludes that the missile penetration will be minimal, with only surface spalling of the concrete.

The Chapter D3.0, Section D3.4.2.6, analysis of the effects of the wind-driven missiles and the bounding evaluation of the effects of the DBT showed that the Rad-Vault is able to withstand any induced stresses without significant damage to the internal containers.

Fire – The TRIGA fuel storage system is analyzed in Chapter D3.0 and summarized in Section D3.4.2.7 against the DBF specified for the 10 CFR 71 transportation fire.

The analysis was conducted to determine the thermal effect of excessive heat on the Rad-Vault storage system, and the internal containers and their contents, which results from a combined maximum temperature solar day and fire.

The analysis assumed equilibrium conditions and determined that radiological material releases will not result from exceeding the temperature limits; however, the design specifications for the exterior concrete material of the Rad-Vault can be exceeded for a short period of time and should be evaluated as part of the recovery action, if a fire event occurs.

D4.4.2.5 Rad-Vault Controls (Technical Safety Requirements).

The assumptions associated with the Rad-Vault that require TSRs to ensure performance of its safety function are as follows:

- The Rad-Vault shall not be moved while loaded or with the lid in place.
- Crane loads other than NRF TRIGA casks, DOT-6M containers, rigging, and the Rad-Vault lid shall not be handled over a loaded Rad-Vault.
- The Rad-Vault lid shall not be handled at a height greater than 12 in. above the top of the Rad-Vault.
- The Rad-Vault lid shall be handled with an approved crane. The Hanford Site hoisting and rigging criteria (DOE/RL-92-36) for a critical lift shall be imposed. A program shall be in place to ensure that only approved cranes are used.
- The Rad-Vault lid shall be replaced after each activity within the Rad-Vault.

- The Rad-Vault shall be positioned on compacted gravel.

D4.4.3 Neutron Radiography Facility TRIGA Cask

The fuel from the 300 Area TRIGA experimental research reactor core/pool storage has been removed as part of the decommissioning of the facility. The TRIGA irradiated fuel inventory consists of 99 TRIGA fuel elements and two FFCRs. The standard fuel elements are stored dry in the NRF TRIGA casks prior to transport to the 200 Area ISA.

D4.4.3.1 Neutron Radiography Facility TRIGA Cask Safety Function.

Based on the results of the Chapter D3.0 consequence analysis, the cask is designated general service; however, the cask provides structural protection of the SNF for several accidents, and as such, is designated as NRC important-to-safety Category B. The general-service NRF TRIGA casks were originally designed and procured in an NRC-equivalent manner. These items are currently in service at the 400 Area ISA. The risk of continued use of the general-service NRF TRIGA casks at the 200 Area ISA is considered acceptable due to the low-activity SNF, as discussed in the Section D3.4.1 analysis section. The NRF TRIGA cask also provides the general-service function to prevent an uncontrolled release of the TRIGA fuel material.

Accidents and concerns identified in the Chapter D3.0 analysis and listed in Table D4-1 and Table D4-2, as associated with the important-to-safety functions for the NRF TRIGA cask, are as follows:

- Handling/drop – Maintain confinement of radioactive materials after a credible drop.
- Cask tipover – Maintain confinement of radioactive materials after cask tipover.
- Fuel rod rupture – Maintain confinement of radioactive materials after rupture of all the fuel pins.
- Seismic – Withstand seismic accelerations without loss of confinement.
- Tornado/wind – Withstand tornado pressure differential without loss of confinement.
- Fire – Withstand transportation DBF within the Rad-Vault without losing confinement or exceeding temperature limits for fuel cladding or container components.

D4.4.3.2 Neutron Radiography Facility TRIGA Cask System Description.

Each NRF TRIGA cask consists of three main components: an inner aluminum basket to hold fuel elements, a surrounding single-wall stainless steel inner container, and an outer stainless steel and lead composite vessel for shielding. There is also an external impact limiter that is not used during storage (WHC-SD-TP-SARP-008).

The inner basket, shown in Figure D2-4, is a welded assembly of machined aluminum tubing and plate 28.5 in. high. Two 1.0-in. thick circular plates, each machined with a hexagonal array of 18 thru-holes, serve as insertion guides and support members for the TRIGA fuel elements. A third aluminum plate, 2.56 in. thick, serves as the fuel seating structure. The top portion of the central aluminum tube has threads that interface with basket-lifting equipment (WHC-SD-TP-SARP-008).

The surrounding stainless steel inner container, shown in Figure D2-4, contributes to the total shielding capability of the cask. The shell of the inner container is 29.12 in. high, with a 10.25-in. outer diameter and an 8.88-in. inner diameter 304 stainless steel tube section. The shell has a machined O-ring groove at the upper face to accommodate a 9.12-in. elastomer seal. The base of the container is a 9.5-in. diameter 304 stainless steel plate that is 0.5 in. thick, which is welded to the lower end of the shell. The lid is a 9.62-in. diameter 304 stainless steel plate that is 1.0 in. thick. The closure system consists of a locking bar and a single stainless steel eyebolt. The locking bar, with a tapped thread at its upper end, is welded to the center of the interior side of the container base. To close, the eyebolt is inserted through the center of the lid and threaded into the upper portion of the locking bar until the proper seal compression is achieved (WHC-SD-TP-SARP-008).

The outer confinement vessel, shown in Figure D2-4, consists of a 12.0-in. outer diameter 304 stainless steel inner shell with a wall thickness of 0.75 in., which serves as the confinement boundary. The 304 stainless steel outer shell has a 16.0-in. outer diameter with a wall thickness of 0.5 in.

Both shells are welded at the lower end to a 4.0-in. thick by 16.0-in. diameter 304 stainless steel plate. The 1.5-in. thick annulus between the two shells is filled with ASTM B29 lead for shielding. The lid is a 16.0-in. diameter 304 stainless steel plate 3.5 in. thick, which has a machined flange that mates with the top of a seal flange using 12 bolts. The thick center portion of the lid fits closely inside of the seal flange and is provided with an O-ring bore seal. The tolerance between the seal flange and the lid are closer than the bolt tolerance. Therefore, during high loading conditions (e.g., a drop), shear is transferred by bearing to the flange. The total height of the outer vessel with lid is 37.7 in. (WHC-SD-TP-SARP-008).

The weight of the outer vessel is approximately 1,637 lb. The entire NRF TRIGA cask weighs a total of 2,013 lb when filled with 18 elements (WHC-SD-TP-SARP-008).

The fabrication materials for the storage system components of the NFR TRIGA cask are as follows:

- Contents basket – ASTM B-210, 6061-T6 aluminum tubing, and ASTM B-209, 6061-T aluminum plate.
- Inner container – ASTM A-511, 304 stainless steel tubing, and ASTM A-240, 304 stainless steel plate.
- Outer vessel – ASTM A-511, 304 stainless steel tubing; ASTM B-29 poured lead; ASTM A-240, 304 stainless steel plate; and ASTM A-540, Grade B23, Class 4 low alloy cap screws.
- Combination Helicoflex O-ring, made of an Inconel spring with an aluminum jacket, with Viton O-ring.
- Ethylene-propylene O-ring bore seal container seals.

- Inconel X-750, Teflon[5]-coated C-ring.
- Swagelok[6] stainless steel quick disconnect with Viton seal.

The confinement boundary for the NRF TRIGA cask consists of the following vessels, lids, and seals:

- Outer vessel closure lid
- Outer vessel, inner shell
- Outer vessel, seal flange
- Outer vessel bottom plate
- Welds on outer vessel, inner shell
- Helicoflex metallic seals.

The outer confinement vessel closure lid uses an elastomeric O-ring bore seal, located in a perimeter groove of the closure lid step, to establish a 10^{-7} scc/sec (air) leak-tight seal within the bore of the cask for initial transport (Figure D2-5). This elastomeric seal has a limited life of approximately 5 years; therefore, a metallic seal is also provided for long-term storage. The Helicoflex metallic seal is located on the surface between the lid and the flange of the outer container. This seal was tested to 10^{-5} scc/sec (air). A leak test port in the lid between the two seals was tested to 10^{-5} scc/sec (air). A Viton redundant seal is also provided to facilitate leak testing of the Helicoflex metallic seal. These seals are embedded and secured to the closure lid and interface with the seal flange of the outer container (Figure D2-5). The Helicoflex seal has a design life of 50 years (WHC-SD-TP-SARP-008), and the Viton seal has a design life of 20 years. A seal verification and replacement program, as described in Section D4.4.3.4, is to be implemented.

Two containment welds are located on the inner shell of the outer vessel. Ultrasonic inspection procedures performed in accordance with the ASME Code, Section V (ASME 1989), are used to inspect the top weld. During fabrication, inspections are performed per the ASME Code, Section VIII, Division 1 (ASME 1989). The bottom weld is radiographically inspected in accordance with the same ASME Code. A vendor cask confinement boundary leak test ensures these welds meet the leak-tight criteria per ANSI N14.5. A pressure test at 17.3 lb/in^2 was also performed.

Another boundary is provided by the inner container, which uses an elastomeric O-ring imbedded in the step of the lid and a metallic seal around the locking eyebolt (Figure D2-5). This boundary is not leak tested, so credit is not taken for it as a confinement boundary; however, the container will enhance fuel retrieval in the future.

Leak tests can be performed on both the elastomeric O-ring bore seal and the Helicoflex metallic seal by the use of three test ports machined on the top side of the lid (Figure D2-5). The test port at Location 1 penetrates the lid into the cask cavity. The port at Location 2 penetrates the lid into the void space between the bore seal and the metallic seal. The port at Location 3 penetrates the annulus of the Helicoflex metallic seal and the outer Viton seal. Each port is fitted with a quick-connect fitting to facilitate a leak test and a cover plate to protect the fittings and

[5] Teflon is a trademark of E. I. du Pont de Nemours and Company.

[6] Swagelok is a trademark of Crawford Fitting Company.

provide shielding equivalent to the lid thickness (WHC-SD-TP-SARP-008). The quick-disconnect fittings are leak tested, and the cover plate's metallic seal is leak tested to 1.0×10^{-5} scc/sec (air) prior to initial transport.

To accommodate the 10 CFR 72 requirement for redundant seals for commercial fuel, credit must be taken for the Viton O-ring used in conjunction with the Helicoflex metallic seal. Since these elastomer seals have a 20-year design life (less than that of the containers), a program will be implemented to verify seal condition and to replace seals, if necessary. A congruent effort to identify long-life seals for replacement of the original seals will also be implemented.

All stresses evaluated for the NRF TRIGA packaging under normal and accident conditions are maintained below the appropriate ASME Code, Section VIII (ASME 1989) allowables. Design, fabrication, and testing of the NRF TRIGA packaging is in accordance with the requirements of the ASME Code, Section VIII, as required by the appropriate NRC Regulatory Guides. Per NRC Regulatory Guide 7.11, *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 in. (0.1 m)*, Table 1, the NRF TRIGA package is Category 3. The design follows the criteria from the ASME Code, Section VIII, Division 2, as shown in the NRF TRIGA SARP, Part B, Section 7.0 (WHC-SD-TP-SARP-008).

Fabrication of the NRF TRIGA cask is in accordance with NUREG/CR-3854. All containment welds were radiographically or ultrasonically tested per ASME Code, Section V. During fabrication, inspections and containment leak testing were performed per the ASME Code, Section VIII, Division 1 and ANSI N14.5, respectively.

D4.4.3.3 Neutron Radiography Facility TRIGA Cask Functional Requirements.

Confinement – The NRF TRIGA cask shall withstand potential NPH and accident events with sufficient integrity to provide a confinement boundary for the TRIGA material such that onsite risk acceptance guidelines are not exceeded.

Handling/Drop – The NRF TRIGA cask shall withstand any induced stresses resulting from the design basis handling/drop accidents with no significant damage and no uncontrolled release of radioactive material.

Cask Tipover – Maintain confinement of radioactive materials after cask tipover.

Fuel Rod Rupture – The NRF TRIGA cask shall withstand the internal pressure from rupture of all of the contained fuel elements with no significant damage and no uncontrolled release of radioactive material.

Seismic – The NRF TRIGA cask, as an integral component of the Rad-Vault system in the 200 Area storage location, shall withstand any induced stresses resulting from the design basis seismic event with no significant damage and no uncontrolled release of radioactive material.

Tornado/Wind – The NRF TRIGA cask, as an integral component of the Rad-Vault system in the 200 Area storage location, shall withstand any induced stresses resulting from the DBT/wind with no significant damage and no uncontrolled release of radioactive material.

Fire – The NRF TRIGA cask, as an integral component of the Rad-Vault system in the 200 Area storage location, shall be capable of withstanding the DBF with no significant damage and no uncontrolled release of radioactive material.

D4.4.3.4 Neutron Radiography Facility TRIGA Cask Evaluation.

Confinement – The NRF TRIGA casks were procured with the intent of meeting the leaktight requirements of 10 CFR 71 for transportation. The containers were fabrication tested by the vendor for a maximum leak rate of less than 10×10^{-7} scc/sec (air). Leak rate criteria for storage of TRIGA fuel in NRF TRIGA casks was established in WHC- TI-75002, *Fast Flux Test Facility Final Safety Analysis Report*, Appendix H, "Fast Flux Test Facility Fuel Offload and Fuel Storage in the Interim Storage Area." Section H.3.4.10.2, "Confinement Requirements" for the NRF TRIGA cask, establishes a maximum storage leak rate of 1.0×10^{-1} scc/s (air) at 1.0 atmosphere differential pressure per ANSI N14.5 (ANSI 1987), based on the low A2 value of the TRIGA fuel. This storage leak rate is bounded by the leak rate requirements for transportation of the NRF TRIGA casks. WHC-SD-TP-SARP-008, Rev 0-C, establishes a maximum leak rate of 6.0×10^{-3} scc/sec (air) for transport of the containers. The NRF TRIGA casks were leak tested upon initial loading according to ANSI N14.5 standards for a maximum leak rate of 1.0×10^{-5} scc/sec (air).

The outer container of the NRF TRIGA cask is the qualified and tested confinement barrier for storage. This container has a Helicoflex metallic seal for long-term storage integrity, an elastomeric flange seal to facilitate leak testing, and an elastomeric bore seal. The quick disconnects used for leak testing are also fitted with metallic O-ring seals at the thread interface. The port lids covering the quick-disconnects used for helium leak testing also are provided with single metallic Helicoflex seals.

Helicoflex seals are fabricated using an inner spring covered by an inner lining and a soft outer lining. The Helicoflex seal for the NRF TRIGA has a Nimonic 90 Spring covered by an inner lining of Inconel Alloy 600 and an outer jacket of aluminum. The materials used in the linings are application-dependent and a function of temperature and pressure. The sealing principle of Helicoflex is based, in part, on plastic deformation of the seal's outer lining. The inner helical spring provides an elastic core under bolt preload. Each coil of the spring behaves independently during radial compression. The all-metal design is the reason for its long life.

The NRF TRIGA cask provides multiple barriers. The inner container has an elastomeric seal on the lid and a metallic seal on the clamping eyebolt. The outer container has redundant seals provided by both an elastomeric bore seal O-ring and a Helicoflex combination metallic/elastomeric O-ring seal. Both the bore seal and the Helicoflex seal are leak tested before transport and/or storage.

The closure devices on the NRF TRIGA cask include the closure lid and the three leak test port lids. The closure lid is attached to the cask body with twelve 0.50-in. diameter cap screws, torqued to a minimum of 52 ± 5 ft-lbf. All leak test port lids are attached to the cask lid with six 0.25-in. diameter cap screws, torqued to a minimum of 42 ± 6 in.-lbf.

The NRF TRIGA cask is provided with removable test port covers that permit access to the helium leak test components in the outer container. The container was leak tested upon initial loading according to ANSI N14.5 standards for a maximum leak rate of 1.0×10^{-5} scc/sec (air), including the test port covers.

The NRF TRIGA cask has a design life of 50 years. All components are stainless steel except the sealed lead annulus and the aluminum basket. There are no neutron poisons. The metallic Helicoflex seal on the outer container has a 50-year design life for this application. The Viton redundant seal has a 20-year design life. The elastomeric bore seal is ethylene propylene and has a 5-year design life. The outer container's elastomeric bore seal is a confinement seal provided to meet transportation requirements. The Helicoflex metallic seal and Viton O-ring are confinement seals provided to accommodate long-term (20-year) storage requirements.

The fuel is stored in untreated air. According to the *American Society for Metals Handbook*, Volume 3 (ASM 1980) and Volume 10 (ASM 1975), the stainless steel and aluminum fuel cladding is not susceptible to long-term degradation in unlimited air. There is no corrosion catalyst within the casks. The reactor pool water was deionized, and the stainless steel casks have been passivated.

The hydride fuel has excellent corrosion resistance in water. Bare irradiated fuel specimens have been subjected to a pressurized water environment at 5,700 °F and 1,230 lb/in^2 during a 400-hour period in an autoclave (Simnad 1981). The average corrosion rate was 350 mg/cm^2/month, accompanied by conversion of the surface layer of the hydride to an adherent oxide film. The maximum extent of corrosion penetration after 400 hours was less than 2 mils. If somehow the cladding was removed from the fuel, the fuel was stored in a corrosive environment (water vapor) within the cask at an elevated temperature, and the cask was pressurized (externally because the fuel would not generate the gas pressure), minimal corrosion will result. For the actual storage configuration: (1) the elements will be dried, ensuring an environment that minimizes corrosion, (2) cladding will be on the fuel, (3) temperatures will not exceed 200 °F, and (4) the resulting maximum pressure will be below the design pressure of 11.2 lb/in^2 gauge. Based on the minimal reactivity of the fuel with water, the design storage environment does not require purified dry air or inert gas.

Dr. Simnad's report (Simnad 1981) and the testing performed on the fuel show there will be no degradation of the fuel, even if the cladding degrades. The fuel itself forms an oxide layer to protect it from corrosion. The environment within the cask is not corrosive for either the cladding or the bare fuel, so the fuel will remain intact. All TRIGA fuel element cladding appears to be intact, as the reactor pool water is not contaminated after 15 years of operation and fuel storage.

Hypothetical degradation of the fuel does not pose unacceptable operational safety considerations with respect to removing the fuel from storage. If repackaging is deemed necessary, it is to be performed in a confinement facility for personnel safety at a yet to-be-determined location.

To accommodate the redundant seal requirement for commercial fuel per 10 CFR 72, credit must be taken for the Viton O-ring used in conjunction with the Helicoflex metallic seal. Since these elastomer seals have a design life less than that of the containers, a program will be implemented to verify seal condition and replace seals, if necessary. An effort to identify long-life seals for replacement of the original seals will also be implemented.

HNF-10094, *NRF TRIGA Cask Seal Plan*, establishes a seal integrity program. The seal integrity program includes assessment, possible replacement, and verification of seals while maintaining a safe worker environment. It should be noted that due to the relatively low source/dose rates of the TRIGA fuel (maximum cask dose of 92 mrem/h contact measured during initial loading, approximately 3 rem shine with cover removed), controlled retrieval and verification may be accomplished in accordance with approved operating procedures in an approved location providing a confinement enclosure. Since the Viton seal has a 20-year design life, this seal integrity program will be implemented prior to 2014.

Drop analyses for the NRF TRIGA cask and the DOT-6M container indicate that they retain confinement and design shielding for the normal handling events and the design basis cask/container drops.

The ethylene propylene seal has a tested life of at least 1,000 hours at 275 °F according to the *Parker O-Ring Handbook* (Parker Hannifin Corporation 1992). The Viton O-ring is rated for a maximum temperature of 400 °F. The maximum normal temperature for the cask is 161 °F due to solar heating of the Rad-Vault, which is well within the design operating range. The Helicoflex metallic seal, fabricated from Inconel and aluminum, has a maximum operating temperature greater than 480 °F. The maximum normal temperature is 161 °F, which is well within the operating range. The maximum accident temperature to which these seals will be exposed results from the DBF when the interior temperature of the Rad-Vault rises to 177 °F. This temperature does not threaten the integrity of the seals due to the large amount of margin between the design limit and analysis results.

The following aspects of TRIGA fuel storage enhance confinement of the fuel:

- Intact cladding (demonstrated)
- Low burnup (cladding not stressed)
- Low fission gas pressure (cladding not stressed)
- Inner containers (not leak tested)
- Leak tested Swagelok fittings
- Leak tested Helicoflex seal (installed per procedure, tested to 1×10^{-5} scc/sec [air]).

These features justify the bolted closure per the approved design. The removable test port covers, as configured, support the seal replacement/demonstration programs required to accommodate 40-year storage.

The inner NRF TRIGA container is fabricated of stainless steel, and the basket is fabricated of aluminum. The 2R inner container of the DOT-6M specification package is constructed of 304L stainless steel, and the support basket is fabricated of aluminum. The fuel cladding is stainless steel or aluminum. There have been reported instances of galvanic corrosion of irradiated aluminum-clad TRIGA fuel stored in stainless steel racks in water. The basket in the NRF TRIGA cask has been specifically designed and fabricated of aluminum to preclude this potential interaction. There are no historical records of stainless steel-clad degradation in aluminum racks. Both types of fuel cladding have been stored in the aluminum fuel storage racks in the 308 reactor, and the reactor core array is also fabricated of aluminum. As such, there is no anticipated interaction between the stainless steel and aluminum metals resulting in deleterious effects on the cladding or container materials.

The TRIGA fuel storage system, composed of the NRF TRIGA casks and DOT-6M containers within the Rad-Vault, can withstand the normal and natural phenomenon conditions anticipated during the 50-year design life of the system. For the DBA identified in Section D3.4.2.2, the Rad-Vault is assumed to fail, allowing damage to the NRF TRIGA casks and DOT-6M containers and resulting in a release of radioactive materials. The offsite and onsite dose consequences are below the release limits and guidelines and also are within limits prescribed in 10 CFR 72 for normal and accident conditions. Although the NRF TRIGA cask and DOT-6M container have the potential to be certified for offsite shipment (they were designed with the intent to meet 10 CFR 71), these containers and the Rad-Vault were all procured as general-service items based on the small radionuclide source term. 10 CFR 72.230 states that casks certified under 10 CFR 71 can be certified as spent fuel storage casks, if they can meet the storage requirements for 20 years. These casks with their metallic seals, in combination with the Rad-Vault, meet the 20-year storage life requirement.

Drop analyses for the NRF TRIGA cask without the impact limiter indicate that the cask retains confinement and design shielding for the normal handling events and the design basis drops. The drop analyses are provided in Section D3.4.2.1.

Under the dynamic conditions of normal transport and hypothetical accidents, the confinement remains sealed.

Handling/Drop – During handling operations, any one or all of the NRF TRIGA casks, DOT-6M containers, or the Rad-Vault could be involved in a drop accident that threatens the confinement or results in significant damage to a storage container. A handling/drop analysis is performed in Chapter D3.0 and summarized in Section D3.4.2.1. The analysis demonstrates that under credible drop conditions, the NRF TRIGA casks and DOT-6M containers do not release radioactive material or are not significantly damaged. It is also noted that the side drop analysis bounds a cask tipover event with no loss of confinement. However, to preclude exceeding the design criteria and the analyzed conditions, lifting limits have been imposed and are identified in Section D3.4.2.1.5 and Chapter D5.0.

Fuel Rod Rupture – The NRF TRIGA cask is analyzed in Chapter D3.0, and the analysis is summarized in Section D3.4.2.4. The TRIGA cask design pressure is based on the maximum pressure differential obtained from the 10 CFR 71.71(c)(3) reduced external pressure. The NRF TRIGA cask is pressure tested at 150% of the design pressure.

As noted in the Chapter D3.0 analysis, the low burnup of the fuel produces a small quantity of fission products. Most of the fission product gases are captured within the fuel matrix. The fission product gases within each element would result in negligible pressure in the cask or container, if all the fuel elements are breached within the cask. Byproducts from elastomeric seal degradation will have minimal effect. Therefore, it is concluded that the NRF TRIGA cask pressure will remain less than the design limit.

As noted in the analysis, the NRF TRIGA package contains stainless steel- and aluminum-clad uranium zirconium hydride fuel. This hydride fuel composition and high equilibrium pressure creates a stable, non-hydrogen-producing fuel. There is no generation of hydrogen gas or corresponding pressure rise inside the NRF TRIGA package during normal conditions. The hydride fuel is assumed to be intact and enclosed within the stainless steel or aluminum cladding. Also, the fuel itself is non-pyrophoric and non-water reactive. Therefore, no chemical reaction will occur that produces gas within the NRF TRIGA package. Also, the fuel is air dried with heat lamps prior to loading. This precludes the possibility of radiolytic decomposition of water.

The analysis also indicates that this fuel forms an oxide film that inhibits the loss of hydrogen or degradation of the fuel. Assuming the fuel cladding disintegrates under accident conditions, the temperatures reached will not exceed the temperature required to generate hydrogen.

The analysis concludes that the maximum temperature of the fuel cladding during storage, if experienced inside the cask or container, will result in an insignificant pressure increase in the fuel.

Seismic – The NRF TRIGA cask is analyzed as an integral component of the Rad-Vault storage system. The analysis summarized in Section D3.4.2.5 shows the design of the Rad-Vault is well within the DBE for the site, and seismic stability is assured. The Rad-Vault will not overturn or slide, and the NRF TRIGA cask will not suffer loss of integrity or loss of confinement resulting in an uncontrolled release of radioactive material.

Tornado/Wind – The NRF TRIGA cask is analyzed for the DBT as an integral part of the Rad-Vault storage system in Chapter D3.0, and the analysis is summarized in Section D3.4.2.6. Wind velocities are bounded by the tornado winds, but the wind missile is also analyzed. A 15-lb., 2x4 traveling at 50 mph only penetrates the concrete surface about 1 in. The tornado winds do not result in sliding or tipover, and there are no induced loads to the cask as a result of the tornado or wind events sufficient to cause a loss of integrity or an uncontrolled release of radioactive material.

Fire – Analysis of the DBF is addressed in Chapter D3.0 and summarized in Section D3.4.2.7. The NRF TRIGA cask is analyzed for the DBF as an integral part of the Rad-Vault storage system. An evaluation is provided in Section D4.4.2.4 for the Rad-Vault system containing the NRF TRIGA casks. Fire protection for the NRF TRIGA cask is provided by the Rad-Vault.

D4.4.3.5 Neutron Radiography Facility TRIGA Cask Controls (Technical Safety Requirements).

The assumptions associated with the NRF TRIGA cask that require TSRs to ensure performance of its important-to-safety function are as follows:

- The maximum handling lift height shall not exceed 109-in. above ground (21 in. above the top of the Rad-Vault with the lid removed).
- The NRF TRIGA cask shall be handled with an approved crane. The Hanford Site hoisting and rigging criteria (DOE/RL-92-36) for a critical lift shall be imposed. A program shall be in place to ensure that only approved cranes are used.

D4.4.4 DOT-6M and 2R Containers

The fuel from the 300 Area TRIGA experimental research reactor core/pool storage has been removed as part of the decommissioning of the facility. The TRIGA irradiated fuel inventory consists of 99 TRIGA fuel elements and two FFCRs. The FFCRs are stored dry in special DOT-6M containers with 2R inner containers, prior to transport to the 200 Area ISA.

D4.4.4.1 DOT-6M and 2R Container Safety Function.

Based on the Chapter D3.0 consequence analysis, the DOT-6M and 2R containers are designated general service; however, the containers provide structural protection of the SNF for several accidents, and as such, are designated as NRC important-to-safety Category B. The general-service DOT-6M and 2R containers were originally designed and procured in an NRC equivalent manner. These items are currently in use at the 400 Area ISA. The risk of continued use of the general-service DOT-6M package and 2R inner containers at the 200 Area ISA is considered acceptable due to the low-activity SNF, as discussed in the Section D3.4.1 analysis section. The DOT-6M package and 2R inner containers also provide a general-service function to prevent an uncontrolled release of the FFCR fuel material.

Accidents and concerns identified in the Chapter D3.0 analysis and listed in Table D4-1 and Table D4-2, as associated with the important-to-safety functions for the DOT-6M container, are as follows:

- Handling/drop – Provide impact absorption for the 2R container.
- Seismic – Provide impact absorption for the 2R container.
- Fire – Withstand transportation DBF conditions within the Rad-Vault such that the 2R container inside does not lose confinement or exceed temperature limits for fuel cladding or container components.

Accidents and concerns identified in the Chapter D3.0 analysis and listed in Table D4-1 and Table D4-2, as associated with the important-to-safety functions for the 2R container, are as follows:

- Handling/drop – Maintain confinement of radioactive materials within the DOT-6M containers after a credible drop.
- Cask tipover – Maintain confinement of radioactive materials after container tipover.
- Fuel rod rupture – Maintain confinement of radioactive materials after rupture of all the fuel pins.
- Seismic – Withstand seismic accelerations without loss of confinement.

- Tornado/wind – Withstand tornado pressure differential without loss of confinement.
- Fire – Withstand transportation DBF conditions within the Rad-Vault without losing confinement or exceeding temperature limits for fuel cladding or container components.

D4.4.4.2 DOT-6M and 2R Container System Description.

Each of the two DOT-6M containers, shown in Figure D2-6, consists of a 16-gauge 304 stainless steel 110-gal drum that is 23 in. in diameter and 69.25 in. high. An inner (2R) vessel is fabricated from a 304 stainless steel pipe with a 5-in. diameter. A 0.5-in. thick stainless steel plate is welded to the bottom of the pipe. The top of the pipe is sealed by a bolted flange assembly containing one Helicoflex metallic seal and one Viton O-ring. The metal seal is leak tested to 1.0×10^{-7} scc/sec (air). The helium test port plug is also leak tested to 1.0×10^{-7} scc/sec (air). This inner vessel is partially supported by plywood treated with a fire-retardant coating. Impact-resistant insulating material fills the annulus between the two vessels according to WHC-S-0393, *Specification for a DOT 6M/2R Metal Packaging*.

The design and fabrication of the DOT-6M outer and inner container were performed in accordance with Title 49, *Code of Federal Regulations*, Part 178, "Specification for Packaging," Sections 49 CFR 178.354 and 49 CFR 178.360, respectively, and the design was approved by DOE. The DOT-6M is a DOT specification packaging approved for offsite shipment as a Type B container. The fabrication of the inner vessel was performed in accordance with the ASME Code, Section VIII, Division 1 (ASME 1989). During fabrication, inspections and containment leak testing were performed per the ASME Code, Section VIII, Division 1, and ANSI N14.5. The weight of the loaded DOT-6M container is approximately 559 lb.

The fabrication materials for the storage system components for the DOT-6M specification package are as follows:

- Drum – Type 304 or 304 L stainless steel.
- Insulation – Celotex[7] fiber insulation in accordance with ASTM C-208; exterior grade A-C fir plywood; Miracle Type-M Black Magic[8] adhesive; PPG-SPEEDHIDE[9] latex 42-7 fire retardant paint; VIAC Mastic WC-5[10] weather coating.
- 2R inner container – ASME SA-240, grade 304L plate and sheet; ASME SA-312, grade TP304L seamless or welded pipe; ASME SA-182, grade F304L flanges; ASME SA-276, Type 304L round bar; ASME B18.8.2, Type 18-8 stainless steel dowel pins.
- Helicoflex metallic seal and Viton O-ring.

When the DOT-6M/2R metal packages were delivered, the 2R materials certification was listed as ASME/ASTM. This equivalency is the result of a long-term joint program by ASTM and ASME. This effort was brought on by the precipitous decrease in nuclear-related

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[7] Celotex is a trademark of the Celotex Corporation.
[8] Miracle Type-M Black Magic is a trademark of Miracle Adhesive Corporation.
[9] PPG Speedhide is a trademark of PPG Industries, Inc.
[10] VIAC Mastic WC-5 is a trademark of VIMASCO Corporation.

construction that made ASME-certified materials almost impossible to obtain in small lots. In many cases, the only difference between ASME steels and identical, more abundant and cheaper ASTM steels has been the testing regime. This availability and cost differential was unnecessarily costly for the nuclear transportation and storage industry; hence, the goal to bring ASME/ASTM specifications for common steel materials into agreement. The result is an ASME specification with annotation similar to the following: ASME SA-240 specification identical with ASTM A-240 ". . . except for editorial differences . . ."

D4.4.4.3 DOT-6M and 2R Container Functional Requirements.

Confinement – The 2R container, as an integral part of the Rad-Vault storage system and as configured with the DOT-6M container, shall withstand stresses induced by potential NPH or accident events at the 200 Area storage location with sufficient integrity to maintain the 2R confinement boundary.

Handling/Drop – The DOT-6M and 2R containers, as integral components of the Rad-Vault storage system in the 200 Area storage location, shall withstand any induced stresses resulting from the design basis handling/drop accidents with no significant damage and no uncontrolled release of radioactive material.

Cask Tipover – Maintain confinement of radioactive materials after cask tipover.

Fuel Rod Rupture – The DOT-6M and 2R container shall withstand the internal pressure from rupture of all of the contained fuel elements with no significant damage and no uncontrolled release of radioactive material.

Seismic – The DOT-6M and 2R containers, as integral components of the Rad-Vault system in the 200 Area storage location, shall withstand any induced stresses resulting from the design basis seismic event with no significant damage and no uncontrolled release of radioactive material.

Tornado/Wind – The DOT-6M and 2R containers, as integral components of the Rad-Vault system in the 200 Area storage location, shall withstand any induced stresses resulting from the DBT/wind with no significant damage and no uncontrolled release of radioactive material.

Fire – The DOT-6M and the 2R inner container, as integral components of the Rad-Vault storage system in the 200 Area storage location, shall be capable of withstanding the DBF and maximum anticipated thermal conditions (including volcanic ashfall) with no significant damage and no uncontrolled release of radioactive material.

D4.4.4.4 DOT-6M and 2R Container System Evaluation.

Confinement – The DOT-6M/2R containers used for storage of the FFCRs provide multiple confinement barriers. The intact cladding is a barrier, and the 2R inner container provides a confinement barrier with the combination metallic and elastomeric O-ring seals. The 2R inner container of the DOT-6M system is the qualified and tested confinement barrier, as discussed in Section D4.4.3.4. This 5-in. diameter pipe vessel has a bolted flange assembly retaining an elastomeric seal and a Helicoflex metallic seal.

Helicoflex seals are fabricated using an inner spring covered by an inner lining and a soft outer lining. The Helicoflex seal for the 2R container has a Nimonic 90 Spring covered by an inner lining of Inconel Alloy 600 and an outer jacket of aluminum. The materials used in the linings are application-dependent and a function of temperature and pressure. The sealing principle of Helicoflex is based, in part, on plastic deformation of the seal's outer lining. The inner helical spring provides an elastic core under bolt preload. Each coil of the spring behaves independently during radial compression. The all-metal design is the reason for its long life.

Although the 2R container was inerted with helium for leak testing, the FFCR fuel can be stored in untreated air. The stainless steel fuel cladding is not susceptible to long-term degradation in unlimited air (ASM 1975 and ASM 1980). There is no corrosion catalyst within the container. The reactor pool water was deionized.

As stated in Section D4.4.3.4, Dr. Simnad's report (Simnad 1981) and the testing performed on the fuel show there will be no degradation of the fuel, even if the cladding degrades. The fuel itself forms an oxide layer to protect it from corrosion. The environment within the cask is not corrosive for either the cladding or the bare fuel, so the fuel will remain intact. All TRIGA FFCR cladding was intact, as the reactor pool water was not contaminated after 15 years of operation and fuel storage.

The DOT-6M stainless steel 6M drum provides a barrier for normal conditions only. This barrier contains pressure relief capability via the penetrations covered with pressure-sensitive adhesive filament (WHC-S-0393). The 6M drum does not provide a safety-related confinement function.

To accommodate the redundant seal requirement for commercial fuel per 10 CFR 72, credit must be taken for the Viton O-ring used in conjunction with the Helicoflex metallic seal. Since these elastomer seals have a design life less than that of the containers, a program will be implemented to verify seal condition and replace seals, as necessary. An effort to identify long-life seals for replacement of the original seals will also be implemented, as discussed in Section D4.4.3.4.

The Viton O-ring is rated for a maximum temperature of 400 °F. The maximum normal temperature for the DOT-6M is 161 °F due to solar heating of the Rad-Vault, which is well within the design operating range. The Helicoflex metallic seal, fabricated from Inconel and aluminum, has a maximum operating temperature greater than 480 °F. The maximum normal temperature is 161 °F, which is well within the operating range. The maximum accident temperature to which these seals will be exposed results from the DBF when the interior temperature of the Rad-Vault rises to 177 °F. This temperature does not threaten the integrity of the seals due to the large amount of margin between the design limit and the analysis results.

The aspects of TRIGA FFCR storage that enhance confinement of the fuel are as follows:

- Intact cladding (demonstrated)
- Low burnup (cladding not stressed)
- Low fission gas pressure (cladding not stressed)
- Leak tested Helicoflex seal (installed per procedure, tested to 1×10^{-7} scc/sec [air]).

The DOT-6M outer stainless steel drum can be opened to permit access to the helium leak test port on the top of the 2R inner container. The 2R vessel was leak tested in accordance with ANSI N14.5 standards for a maximum leak rate of 1.0×10^{-7} scc/sec (air). This test port closure was also leak tested to 1×10^{-7} scc/sec (air).

The FFCR storage system, composed of the DOT-6M containers within the Rad-Vault, can withstand the normal and natural phenomenon conditions anticipated during the design life of the system. Although the DOT-6M container could be certified for offsite shipment, these containers and the Rad-Vault were all procured as general-service items based on the small radionuclide source term. 10 CFR 72.230 states that casks certified under 10 CFR 71 can be certified as spent fuel storage casks, if they can meet the storage requirements for 20 years. These containers with their metallic seals, in combination with the Rad-Vault, meet the 20-year storage life requirement.

Handling/Drop – The DOT-6M and 2R containers are analyzed in Chapter D3.0, and the analysis is summarized in Section D3.4.2.1. Tests were performed on DOT-6M shipping packages (with 2R inner containers) to evaluate their response to a drop onto an unyielding surface and a drop onto a steel punch bar. A detailed observation of the packages following the drop testing showed the security bolts to be tight and no damage to other areas that would indicate the possible loss of confinement. It is also noted that the side drop analysis bounds a cask tipover event with no loss of confinement.

As indicated in the analysis, the only differences between the packages tested and the TRIGA FFCR configuration are the type of seal used in the 2R inner container and the anticipated contents within the packages. The drop-tested packages used an elastomeric O-ring seal, whereas the TRIGA FFCR DOT-6M shipping packages will use a metallic O-ring seal to assure long-term storage requirements. In addition, the drop-tested package used can and bottle internal containers, whereas the TRIGA FFCR DOT-6M shipping packages will contain stainless steel clad fuel rods enclosed in stainless steel pipe. The TRIGA FFCRs are not externally contaminated and the solid hydride fuel itself is enclosed within the stainless steel cladding, which has been demonstrated not to leak.

Based on the results of the analysis, the loaded DOT-6M container has an administrative restriction to equal the NRF TRIGA cask maximum lift height and shall not be handled at a height greater than 21 in. above the top of the Rad-Vault (with the lid removed).

Fuel Rod Rupture – Fuel rod rupture is analyzed in Chapter D3.0, and the analysis is summarized in Section D3.4.2.4. As noted in Section D.4.4.3.4, the low burnup of the fuel produces a small quantity of fission products. Most of the fission product gases are captured within the fuel matrix. The fission product gases within each element will result in negligible pressure, if the fuel element is breached within the 2R container.

The Chapter D3.0 analysis indicates that the DOT-6M package contains stainless steel-clad uranium zirconium hydride fuel. This hydride fuel composition and high equilibrium pressure creates a stable, non-hydrogen producing fuel. There is no generation of hydrogen gas or corresponding pressure rise inside the DOT-6M/2R package during normal conditions. The analysis also indicates that this fuel forms an oxide film that inhibits the loss of hydrogen or degradation of the fuel. Assuming the fuel cladding disintegrates under accident conditions, the temperatures reached will not exceed the temperature required to generate hydrogen. The analysis concludes that the maximum temperature of the fuel cladding during storage, if experienced inside the cask or container, results in an insignificant pressure increase in the fuel.

Seismic – The DOT-6M and 2R containers are analyzed as an integral component of the Rad-Vault storage system. The analysis in Section D3.4.2.5 shows the design of the Rad-Vault is well within the DBE for the site, and seismic stability is assured. The Rad-Vault will not overturn or slide, and the containers will not suffer loss of integrity or an uncontrolled release of radioactive material. Seismic accelerations are bounded by the drop analyses.

Tornado/Wind – The DOT-6M and 2R containers are analyzed for the DBT as an integral part of the Rad-Vault storage system in Chapter D3.0, and the analysis is summarized in Section D3.4.2.6. Wind velocities are bounded by the tornado winds, but the wind missile is also analyzed. A 15-lb., 2x4 traveling at 50 mph only penetrates the concrete surface about 1 in. The tornado winds do not result in sliding or tipover, and there are no induced loads to the containers as a result of the tornado or wind events sufficient to cause a loss of integrity or an uncontrolled release of radioactive material.

Fire – The DOT-6M and 2R containers are analyzed for the DBT as an integral part of the Rad-Vault storage system in Chapter D3.0, and the analysis is summarized in Section D3.4.2.7. Sufficient fire protection for the DOT-6M and 2R container is provided by the Rad-Vault. Exposure to the DBF or maximum thermal conditions resulted in no loss of integrity or release of radioactive material.

D4.4.4.5 DOT-6M and 2R Container Controls (Technical Safety Requirements).

The assumptions associated with the DOT-6M and 2R containers that require TSRs to ensure performance of their important-to-safety function are as follows:

- The loaded DOT-6M container shall not be handled at a height greater than 21 in. above the top of the Rad-Vault (with the lid removed). (Administrative restriction to equal NRF TRIGA cask maximum lift height of 109-in.)
- The DOT-6M container shall be handled with an approved crane. The Hanford Site hoisting and rigging criteria (DOE/RL-92-36) for a critical lift shall be imposed. A program shall be in place to ensure that only approved cranes are used.

D4.4.5 NAC-1 Cask System

The safety-significant functions for individual components of the NAC-1 cask system are discussed as an integral part of the NAC-1 cask system evaluation. The components include the inner canister, the NAC-1 cask, and the ISO shipping/storage container. Specific individual component evaluations are addressed as appropriate.

The 300 Area LWR fuel consists of irradiated commercial LWR fuel provided to the Pacific Northwest National Laboratory Material Characterization Center. The Material Characterization Center was responsible for conducting laboratory investigations of nuclear waste forms for the DOE Office of Civilian Radioactive Waste Management program. The

300 Area LWR fuel inventory consists of seven commercial fuel assemblies, five PWR and two BWR fuel assemblies, and 26 individual intact fuel rods and miscellaneous segments.

The 300 Area LWR fuel is to be contained in welded inner canisters (Figure D2-14) placed within NAC-1 or Nuclear Fuel Services (NFS)-4 spent fuel shipping casks that satisfy both onsite transportation and storage requirements. The NAC-1 and NFS-4 casks were fabricated to the same design drawings, but at different times by different corporate owners. NAC purchased the NFS-4 design and fleet of casks from NFS and renamed the cask model NAC-1. Both model casks will be referred to as the model NAC-1 cask throughout this document.

The 300 Area LWR fuel storage system, composed of the NAC-1 cask, inner canister, and an ISO shipping/storage container, can withstand the normal and natural phenomenon conditions anticipated during the 50-year design life of the system. As identified in Chapter D3.0 and evaluated in Section D3.4.2, the LWR fuel storage system survives all DBAs without loss of confinement of the inner canister, and therefore allows no radiological release under normal or accident conditions. 10 CFR 72.230 states that casks certified under 10 CFR 71 can be certified as spent fuel storage casks, if demonstrated in a SAR that the casks can meet the storage requirements for 20 years. The NAC-1 cask and the inner canister with welded closures exceed the 20-year storage life requirement.

D4.4.5.1 NAC-1 Cask System Safety Function.

The NAC-1 cask system provides passive protection from stresses resulting from natural phenomena and accident events that could compromise the integrity of the inner canister and fuel cladding and allow an uncontrolled release of radioactive material in excess of onsite guidelines. The NAC-1 cask is a passive barrier that has been designated safety significant for the mitigation of potential consequences resulting from NPH and handling events.

The LWR canister is designated safety significant in that it provides a confinement function in the event of pressurization that results from failure of the fuel cladding during the storage life of the LWR fuel. The function is to maintain integrity sufficient to preclude uncontrolled release of radioactive material. It is also designated safety class for criticality geometry control and maintaining a leaktight configuration, as discussed in Section D4.3.2.

Because of its configuration, the ISO container is designated safety significant as part of the NAC-1 cask system in that it provides a passive function in preventing tipping or sliding during accident or NPH events. The function does not require the outer panels of the ISO to remain intact following the design basis event. Neither the ISO container nor the NAC-1 cask has a confinement function.

DBAs identified in the Chapter D3.0 analysis and listed in Table D4-1 and Table D4-2, as associated with the safety-significant functions for the NAC-1 cask, are as follows:

- Handling/drop – Provide passive protection of the LWR canister such that it retains structural integrity after a credible drop.
- Mobile crane fall – Provide passive protection of the LWR canister such that it maintains confinement after a crane fall accident.

- Cask tipover – The NAC-1 cask in the ISO container is credited with passive design features that preclude tipping.
- Seismic – Withstand seismic accelerations without tipover or sliding.
- Tornado/wind – Withstand DBT winds (excluding DBT missiles). Also withstand the design basis wind and wind-driven missiles established for the 200 Area ISA, without structural damage to the LWR canister.
- Fire – Withstand transportation DBF conditions such that the LWR canister does not lose confinement or exceed temperature limits for fuel cladding or cask components.

DBAs identified in the Chapter D3.0 analysis and listed in Table D4-1 and Table D4-2, as associated with the safety-significant functions for the LWR canister, are as follows:

- Handling/drop – Maintain confinement of radioactive materials after a credible drop within NAC-1 cask.
- Fuel rod rupture – Maintain confinement of radioactive materials.
- Seismic – Withstand seismic accelerations without loss of confinement.
- Tornado/wind – Withstand tornado pressure differential without loss of confinement.
- Fire – Withstand transportation DBF conditions inside the NAC-1 cask without loss of confinement.

DBAs identified in the Chapter D3.0 analysis and listed in Table D4-1 and Table D4-2, as associated with the safety-significant functions for the ISO, are as follows:

- Seismic – Provide structural support to the NAC-1 cask to withstand seismic accelerations without tipover or sliding.
- Cask tipover – The NAC-1 cask in the ISO container is credited with passive design features that preclude tipping.
- Tornado/wind – Provide structural support to the NAC-1 cask to withstand tornado winds without tipover or sliding.

D4.4.5.2 NAC-1 Cask System Description.

The 300 Area LWR fuel storage system consists of six storage units, each comprised of an inner canister, a NAC-1 cask, and an ISO shipping/storage container. The welded inner canister provides confinement of the LWR fuel during storage, as required by 10 CFR 72. The NAC-1 provides a shielding overpack that provides weather, accident, and NPH protection.

The NAC-1 casks, previously licensed by the NRC to transport LWR spent fuel and waste material, will be modified for use at Hanford. Modifications to the cask include the removal and plugging of several valves connected to the cask cavity. The NAC-1 casks are mounted to supports within the ISO container for transportation and will remain in the containers in this configuration during storage at the 200 Area ISA.

Additional modifications to the cask include removal of the anti-rotational lugs within the interior cask cavity to accommodate the spent fuel LWR canister. Also, neutron shield tank pressure relief penetrations have been removed and replaced with threaded solid plugs.

The storage units will be placed by crane alongside each other (1 x 6 array) and evenly spaced 4 ft apart. This spacing is not required for criticality or other safety analysis purposes, but rather for personnel access considerations and maintaining personnel radiation doses ALARA.

The NAC-1 cask is designed and fabricated to the requirements of the ASME Code, Section III, Subsection NB. The external shape of the NAC-1 spent fuel shipping cask approximates a smooth-surface, right circular cylinder that is modified, in that impact limiters protrude radially at both ends (see Figure D2-10). The internal cross-section of the cask cavity is circular. The overall dimensions of the cask include a length of 214 in. (including lid impact limiter) and a maximum cross-sectional envelope diameter of 50 in. The internal cavity of the cask is 178 in. long and 13.5 in. in diameter. The maximum loaded gross weight of the cask, including the maximum fuel and inner canister weight (3,300 lb), is approximately 47,150 lb. The principal design features of the cask are the structure, shielding and heat dissipation systems, and the lifting and tie-down systems.

The NAC-1 cask provides shielding and environmental protection of the inner LWR canister. The structures of the NAC-1 cask are constructed of stainless steel. The cask cavity is formed by the inner shell, which is a 14.125-in. outside diameter, 0.3125-in. thick stainless steel shell. The upper end of the shell is welded to the cask cavity flange; the bottom end of the shell is welded to the cask bottom casting. Surrounding the inner shell of the cask is a nominal 6.6875-in. thick annulus of chemical-grade lead (gamma) shielding. The lead is shaped such that approximately 5 in. from the bottom and 30 in. from the top, the thickness is reduced to 5.4375 in. There is an annular void, 5 in. long by 1.25 in. thick, at the bottom end of the gamma shield to allow for any lead expansion during the fire accident. The upper axial shaping is accomplished by reducing the diameter of the outer shell 2.50 in. over a 30-in. length. The lead/steel interface of the inner and outer shell has axial copper fins that are imbedded in the lead and welded to the inner and outer shells to transfer heat across the interface with a minimum temperature gradient (see Figure D2-10).

The outer shell is formed by a 30-in. diameter, 1.25-in. thick stainless steel cylinder reduced to a 27.50-in. diameter at one end. The cask bottom consists of a shaped stainless steel disc with a 30-in. outer dimension, 2 in. thick between the inner and outer shells, and an 8-in. thickness that functions as a gamma shield for the bottom end of the cask. The cavity flange is a stainless steel ring with a 29.75-in. outer dimension, 17.50-in. inner dimension, and an 8.625-in. thickness. The bottom disc end and top flange are welded to the inner and outer shells to form the enclosure for the lead gamma shield.

The primary structure of the NAC-1 cask is the inner shell, lower end casting, upper end casting, bolted closure lid with double O-ring seals, the plugged drain valves and rupture disk penetrations, and the vent/helium fill valve that remains operable. Credit is taken for protection of the canister by the structural characteristics of the NAC-1 cask to demonstrate no loss of confinement during design basis events and accidents.

The inner shell of the NAC-1 cask is a 0.3125-in. thick, austenitic stainless steel right-circular cylinder with internal dimensions of 13.50 in. in diameter and 178 in. in length. At each end, the shell is welded to large austenitic stainless steel end castings. The lower end casting is 3.50 in. thick, with a central region that is an 8-in. thick frustum of a cone. The upper end cask cavity flange is an 8-in. thick annular casting with a tapered central hole that mates with the closure lid. The inner shell is joined to the austenitic stainless steel upper and lower end castings using circumferential full penetration groove welds. All weld configurations are designed and fabricated to meet the ASME Code, Section III (ASME 1971).

The closure lid is a 7.5-in. thick austenitic stainless steel frustum of a cone with a 2-in. thick flange around its circumference at the larger end. A 0.375-in. x 45° chamfer on the lower outer edge of the lid flange is optional; it has no significant effect on the lid's structural adequacy. The closure lid is retained by six, equally spaced, 1-1/4-7UNC ASTM A-320, Grade L43, low alloy steel hex-head capscrews and is sealed at the upper end casting by two polytetrafluoroethelene O-ring seals.

The two drain valves located in the lower end casting are plugged for the storage configuration.

The pressure relief system located in the upper end casting is plugged for the storage configuration, since no postulated storage pressurization events exceed the design pressure of the cask (65 lb/in^2). Pressures within the NAC-1 cask remain below design pressure limits for both normal and accident conditions.

There is one other penetration in the wall of the upper end casting. This penetration was used as a test port for the region between the O-ring seals.

Existing cover plates over cask confinement penetrations will remain in place, as designed, to serve as heat shields during the postulated DBF accident.

Neutron shielding tanks are provided in the 4.5-in. thick annular space formed between the outer shell of the lead gamma shield and a thin stainless steel shell that constitutes the outer cask surface. The neutron shield tanks are not used for the storage configuration at the 200 Area ISA and will contain air.

Upper end shielding is provided by the 7.5-in. thick stainless steel cask lid. The cask lid is a stainless steel casting that also serves as a gamma shield. The lid is a flanged frustum of a cone, 7.5 in. thick with a maximum diameter of 25.5 in. The conical portion of the stainless lid is stepped locally to minimize the gap between the lid and cavity flange, preventing displacement of the lid during the design basis drop. The flanged portion of the lid is a 2-in. thick, 25.5-in. diameter disc. There are six counterbored clearance holes for the lid bolts and four 1-in. diameter blind-threaded holes for attaching the lid impact limiter. The underside of the lid flange has a groove for the two O-ring seals that seal the cask cavity. The upper face of the cavity flange is machined flat to serve as the sealing surface for the cask lid seals. The cask lid is bolted to the cavity flange by six 1.25-in. diameter hex-head bolts. Bolt heads bear on the cask lid; the shanks penetrate through the lid flange and thread into the cavity flange. Six 1.25-in. diameter holes with HeliCoil thread inserts are provided in the cavity flange for bolting the cask lid to the flange. The bolt heads are drilled for a wire security seal.

The lower end impact limiter structure is a ring that surrounds the cask lower casting, formed from a stainless steel sheet and/or plate welded to the cask outer shell and flange areas. The impact limiter was designed to absorb the energy of the design basis 30-ft free drop. It contains a balsa wood disc placed adjacent to the cask bottom. A 0.125-in. thick sheet of asbestos is placed between the balsa and the cask bottom, and is contained internally. Within the impact limiter, extending radially from the center of the cask, are eight 0.375-in. thick stainless steel gussets. The bottom section of the impact limiter also functions as a pedestal for supporting the cask in the vertical position.

An impact limiter located at the upper end of the cask body is designed to absorb the energy of the design basis side drop accident. The upper impact limiter is a stainless steel-sheathed, balsa-filled ring that surrounds the cask cavity flange. A 0.125-in. sheet of asbestos is positioned between the balsa and cask outer shell, and is contained internally. Within the impact limiter, extending radially from the center of the cask, are eight 0.375-in. thick stainless steel gussets. The upper impact limiter, also a cask support member in the storage configuration, rests in the ISO cradle frame cross-member.

The cask lid is protected from impact by a 12-in. thick, balsa-filled lid impact limiter that covers and overlaps the cask lid and cavity flange. The balsa is enclosed within a 0.109-in. thick stainless steel container. A 0.125-in. sheet of asbestos is positioned between the balsa and the sheet material adjacent to the cask, and is contained internally. The impact limiter is attached to the cask lid by four 1-in. diameter bolts. There are elastomer O-rings in grooves under the heads of the 1-in. bolts and a neoprene gasket on the perimeter of the impact limiter. These impact limiter seals are for weather protection and do not provide a confinement function. Removal of the lid impact limiter allows access to the cask lid.

Lifting devices for the NAC-1 cask are designated as lifting trunnions and rotation trunnions. The lifting trunnions are two 8.625-in. diameter by 3-in. long trunnions located on the perimeter of the upper impact limiter. The cask is lifted by a special handling yoke attached to the two trunnions. The rotation trunnions are two 6.625-in. diameter by 3-in. long trunnions for rotating the cask to and from the horizontal position in the ISO container. The lower trunnions are offset from the cask centerline so that when the cask is lowered into the ISO container, it rotates to a horizontal position as the crane hook is lowered.

Transportation of the NAC-1 cask is in the normal storage configuration with the cask in a horizontal position, secured within the ISO container. Two structural cross-member sections serve as cradles for the cask within the ISO container. The lower (rotation) trunnions of the cask are captured by a notch and clamping plate on the aft cradle. The upper lifting trunnions are also captured by clamping plates, to hold the upper end of the cask as it rests on the impact limiter within a neoprene-lined forward cradle. The lid impact limiter is bolted to the cask lid. The lid impact limiter can be unbolted and rolled away from the cask on a track. The cask and ISO container are lifted by crane from the transport trailer and placed in the storage array on the NAC-1 reinforced concrete storage pad.

The NAC-1 cask contains an inner LWR canister that provides a confinement boundary for the fuel. The inner canister is fabricated from a 12-in. stainless steel pipe with welded base cap and top closures. It has a nominal outside diameter of 12.75 in., with a maximum outside diameter of 13.375 in. at the end cap and a total length of 177.36 in. The bottom cap is machined, so the pipe-to-cap weld is inspectable at the side of the canister. The closure lid contains a penetration that allows the canister to be evacuated, filled with helium, and leak tested to the requirements of ANSI N14.5. Figure D2-14 provides an illustration of the PWR inner canister.

The LWR canisters are designed to provide criticality geometry control such that a loaded canister will have a k_{eff} less than 0.95 when fully moderated and reflected. Criticality is incredible if flooding within the canister is precluded, crediting the leaktight configuration of the LWR canister as an independent feature (SNF-8924, *CSER 01-011: Criticality Safety Evaluation for Light Water Reactor Fuel in NAC-1 Casks*).

The design weight of the inner canister is 1,250 lb. The maximum loaded weight of the canister is not to exceed 3,300 lb. The maximum internal design pressure of the canister is 75 lb/in^2 gauge.

A consolidated fuel rod container fits within the inner canister to hold individually packaged LWR fuel rods. The consolidation container is an 8-in. by 8-in. box that is 168 in. long. The lid of the box is hinged and secured closed with 24 screws. The box is closed on the ends with perforated plates to allow helium flow during evacuation, drying, and backfill operations. Confinement of the radioactive material is provided by the welded LWR canister.

The NAC-1 cask is transported and stored within a specially designed ISO shipping container. The ISO container provides weather protection for the cask, which is not a safety function. The ISO containers were fabricated in two heights, in both 6-ft and 8-ft high models. All containers are painted carbon steel construction.

The 8-ft ISO container, fabricated by Evergreen Heavy Industrial Corporation, is nominally 8 ft high x 8 ft wide x 20 ft long. The container has single full-width doors at each end and a removable roof. The frame of the container is structural steel channel construction with a 12.7 mm carbon steel floor. The sides are fabricated of 1.6 and 2.0 mm corrugated carbon steel. The roof is fabricated of 1.6 mm carbon steel sheet metal supported by 40 x 40 mm angle iron on a 60-cm grid and is slightly pitched to prevent ponding of precipitation. The doors and roof are provided with weather seals. Two of these containers will be used to house NAC-1 casks during fuel transport and storage.

The 6-ft ISO container, fabricated by Adamson Containers Ltd., is 6 ft high x 8 ft wide x 20 ft long. The container has two half-width doors at one end and a removable roof. Materials of construction and dimensions are similar to those used in fabricating the 8-ft containers, except the roof material is 1.4 mm carbon steel sheet metal and the roof is flat. Four of these containers will be used to house NAC-1 casks during fuel transport and storage.

The containers have been repainted white to produce lower surface temperatures for industrial safety during their storage in direct sunlight at the ISA. The roof panels are epoxy coated to provide enhanced weather protection.

D4.4.5.3 NAC-1 Cask System Functional Requirements.

Individual components of the NAC-1 cask system are analyzed in Chapter D3.0 as integral parts of the NAC-1 cask. The evaluation of the DBAs is discussed in Section D4.4.5.4. Specific individual component functional requirements and evaluations are addressed as appropriate.

Confinement – The LWR canister shall provide confinement of the LWR fuel. Confinement, as required by 10 CFR 72, is provided by the LWR canister. The double welded LWR canister shall ensure that no uncontrolled release of radioactive material occurs.

Handling/Drop – The NAC-1 cask system shall withstand any induced stresses resulting from the design basis handling/drop accidents with sufficient integrity to act as a passive barrier and prevent damage to the inner canister. The LWR canister, analyzed as an integral part of the NAC-1 cask/container system, shall be capable of withstanding any induced stresses resulting from design basis handling/drop events with sufficient integrity to ensure that confinement of the fuel material is maintained, structural integrity is maintained for criticality geometry control, and the leaktight configuration is maintained to preclude water ingress.

Mobile Crane Fall – The NAC-1 cask system shall withstand any induced stresses resulting from accidents involving the crane, with sufficient integrity to act as a passive barrier and prevent damage to the safety-class LWR canister. The NAC-1 cask shall protect the LWR canister to ensure no uncontrolled release of radioactive material occurs.

Cask Tipover – The NAC-1 cask in the ISO container is credited with passive design features that preclude tipping. The NAC-1 cask in the ISO container shall be analyzed to not tip over or slide as a result of NPH or DBAs.

Fuel Rod Rupture – The LWR canister, as an integral component of the NAC-1 cask system, shall withstand the internal pressure resulting from rupture of all contained fuel rods without compromising the canister structural integrity required to maintain confinement of the fuel material and its leaktight configuration. The design pressure of the LWR canister is 75 lb/in^2 gauge. The LWR canister shall ensure that no uncontrolled release of radioactive material occurs.

Seismic – The NAC-1 cask system shall withstand any induced stresses resulting from the design basis seismic event with sufficient integrity to act as a passive barrier and prevent damage to the safety-class LWR canister. The NAC-1 cask and the ISO container shall protect the LWR canister to ensure no uncontrolled release of radioactive material occurs. The LWR canister, analyzed as an integral part of the NAC-1 cask/container system, shall be capable of withstanding any induced stresses resulting from the design basis seismic event with sufficient integrity to ensure that confinement of the fuel material is maintained, structural integrity is maintained for criticality geometry control, and the leaktight configuration is maintained.

Tornado/Wind – The NAC-1 cask system in the 200 Area storage location shall be capable of withstanding the DBT/wind with sufficient integrity to act as a passive barrier and prevent damage to the safety-class LWR canister. The NAC-1 cask and the ISO container shall protect the LWR canister to ensure no uncontrolled release of radioactive material occurs. The LWR canister, analyzed as an integral part of the NAC-1 cask/container system, shall be capable of withstanding the DBT/wind and pressure differential with sufficient integrity to ensure that confinement is maintained along with the leaktight configuration to preclude water intrusion.

Fire – The NAC-1 cask system in the 200 Area storage location shall be capable of withstanding the DBF with sufficient integrity to act as a passive barrier and prevent damage to the LWR canister. The LWR canister, as an integral part of the NAC-1 cask system, shall be capable of withstanding the DBF with sufficient integrity to ensure that confinement is maintained. The zircaloy fuel cladding has a maximum cladding temperature limit of 644 °F to ensure that failure of the cladding material does not occur. The LWR canister shall ensure that no uncontrolled release of radioactive material occurs, structural integrity is maintained for criticality geometry control, and the leaktight configuration is maintained.

D4.4.5.4 NAC-1 Cask System Evaluation.

Confinement – Confinement of the spent fuel is provided by the leak-tight welded LWR canister, as described in Section D4.3.2.2. The welded canister ensures that the inert gas within the canister cavity following leak testing is maintained during the storage of the LWR fuel. In storage, the NAC-1 cask is not credited for confinement. The LWR canister provides the confinement boundary to satisfy the confinement and retrieval requirements of 10 CFR 72. The NAC-1 cask is a significant structural system that mitigates most normal, off-normal, and accident loads and conditions. It is this structure that is analyzed for most scenarios in this section. The integrity of the LWR canister remains intact for the bounding accidents analyzed in Section D3.4.2 for each category due to the design of the NAC-1 cask, and no uncontrolled release of radioactive material occurs. The cask provides a passive safety-significant function with regard to potential consequences resulting from NPH and accident-induced stresses.

Handling/Drop – These analyses are addressed in Chapter D3.0, summarized in Section D3.4.2.1, and discussed in the LWR canister safety-class evaluation in Section D4.3.2.4. As indicated in the safety-class evaluation, the calculations considered all components as part of an integral cask/container system.

The cask analyses evaluated to hypothetical accident conditions (10 CFR 71.73) included (1) a free drop of the cask through a distance of 30 ft onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected, followed by (2) a free drop of the cask through a distance of 40 in., in a position for which maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar mounted on an essentially unyielding, horizontal surface.

The postulated damage that is incurred by the NAC-1 cask during the hypothetical 10 CFR 71 accident is minimal and does not diminish the ability of the cask to protect the LWR canister to maintain the canister confinement boundary. Any deformation will be highly localized, and the bending stresses in the remainder of the cask will not result in failure of the internal stainless steel shells.

The NAC-1 cask system and the LWR canister are able to withstand any induced stresses from handling/drop events without compromising the integrity or the confinement function of the LWR canister.

Mobile Crane Fall – This evaluation is provided in Chapter D3.0 and summarized in Section D3.4.2.2. As indicated, the calculations consider all components as part of an integral cask/containment system. The accident analysis postulates that the crane boom drop may be sufficient to breach the cask. The Chapter D3.0 analysis determines that the crane boom will fail before the boom can impart sufficient load on the cask to breach confinement. The analysis also investigates whether the cask could be punctured by the crane or other related lifting and rigging hardware. This analysis determines that although the cask will suffer physical damage, the confinement integrity will be maintained in the maximum crane block lift condition. The maximum crane block height that can be achieved by the crane is 80 ft above the cask.

Analysis of a hydraulic spreader bar ISO lifting fixture drop (Section D3.4.2.2) concludes that bending is the controlling parameter that establishes the requirement for a maximum lift height. Although the cask may suffer physical damage, the confinement integrity of the cask will be maintained by imposing a maximum height condition of 6 ft above the ISO container lid, approximately 14.75 ft above the ground (see Chapter D5.0 for load height restrictions). This height provides sufficient overhead clearance to perform all rigging and hoisting activities at the ISA.

The cask system, with the imposition of the maximum lift height restriction, will withstand any induced stresses from a mobile crane fall without impacting the integrity of the LWR canister.

Fuel Rod Rupture – This analysis is addressed in Chapter D3.0 and summarized in Sections D3.4.2.4 and D3.4.2.7. The analyzed hypothetical accident scenario postulates the rupture of all fuel pins of a worst-case hypothetical PWR assembly or the consolidated BWR and PWR individual rods under both normal and fire accident conditions, which causes an increase in cavity pressure that results in possible loss of fuel confinement. The calculation of the maximum normal pressure for the LWR canister assumes a normal cask cavity pressure at the temperature resulting from the hottest summer day (115 °F ambient) added to the pressure increase from fuel rod rupture of all rods. The pressure increase within the cask cavity is due to both a temperature increase from the indoor loading temperature at the 324 Building (68 °F) to the maximum normal cavity storage temperature (372 °F) and 100% rod rupture. The evaluation addresses the fission gas generated during irradiation, initial pressurization within the pins, and the gas within the canister cavity from loading and leak-testing activities.

The pressure within the cavity due to rod rupture at maximum normal conditions is within the design pressure of 75 lb/in^2 gauge established for the LWR canister. Worst-case pressures are also calculated for 100% rod rupture during the DBF conditions. The resultant pressure at this increased temperature is also well within the design pressure for the LWR canister.

The presence of free water in the canister is considered incredible since the fuel is loaded dry, the canister is vacuum dried and backfilled with helium, the closure is double welded, and the canister is leak tested to leaktight standards. This leaktight configuration is maintained during DBA conditions by protecting the confinement function and the structural integrity of the canister.

The LWR canister is able to withstand any induced stresses from fuel rod rupture events without compromising the confinement function of the LWR canister.

Seismic – The seismic evaluation is provided in Chapter D3.0, and the analysis is summarized in Section D3.4.2.5. The NAC-1 cask in the ISO container is shown to not tip over or slide as a result of seismic forces. As noted, stresses induced in the NAC-1 cask and the LWR canister during the 200 Area ISA DBE are within the cask system design capabilities. The LWR canister safety-class evaluation calculations consider all components to be part of an integral system.

Seismically induced loads are bounded by the drop impact accelerations, the evaluation of which concludes that the cask system is able to withstand any induced stresses from handling/drop events without compromising the confinement function of the LWR canister.

Tornado/Wind – The analysis of wind exposure effects of the DBT and wind-driven missiles is provided in Chapter D3.0 and summarized in Section D3.4.2.6. The analysis considered all components as integral parts of an LWR storage system.

The ISO shipping containers are provided in two heights: 6 ft and 8 ft. Since the 8-ft high container has more area exposed to the tornado wind, it was examined as the critical case for wind exposure effects (tipping and sliding). The analyses show that the cask and shipping container will not tip or slide as a result of the DBT (sum of rotational and translational velocities). Tipping of the cask/container is resisted by the force developed as a result of the dead weight of the cask/container. Sliding of the cask/container is resisted by the frictional force between the container and ground.

It is anticipated that the shipping container might sustain considerable damage during the tornado event that results in possible collapse of the side walls and dislocation of the roof sheeting. Thus, the Chapter D3.0 analyses conclude that the shipping container may require inspection and repair following a tornado event, but the cask will endure the wind effects of the DBT. For the wind effects analysis, it is conservatively assumed that the container walls and roof structure remained intact, since this is the condition that imposes the greatest resultant forces relative to tipping and sliding the fuel storage cask. Because external damage to the ISO box was assumed in the analysis, the ISO container side, end, and roof panels are not required for structural integrity and are therefore considered general service.

The LWR canister provides confinement for the stored fuel, but the cask inner shell is the component that must withstand the resultant pressure transient sustained during the DBT event, assuming the ISO container roof and walls collapse. The analysis indicates that the cask inner shell will withstand the pressure transient criteria for the DBT with large margins of safety, and therefore, the NAC-1 cask will endure pressure transient effects of the DBT. The LWR canister design pressure is much greater than the 0.9 lb/in^2 gauge pressure differential of the DBT.

As noted above, the tornado wind is capable of causing significant physical damage to the shipping container, which will require inspection and repair or possibly repackaging of the stored fuel following a tornado event.

The loads resulting from the design basis wind are bounded by the consequences of the DBT discussed above, with the exception of wind-driven missiles. The wind-driven missiles are smaller and less energetic than the analyzed tornado missile, but have a smaller effective area of impact and require separate evaluation. The consequences of tornado missiles for the NAC-1 cask storage are not required to be analyzed, as noted above. However, the analysis is performed in the NAC-1 SAR (NAC-E-804) and is judged to have insufficient energy to perforate, puncture, or cause structural damage to the NAC-1 cask that is sufficient to cause the loss of the confinement function. The analysis, summarized in Section D3.4.2.6, also determines the minimum thickness required to prevent perforation by a wind missile. The evaluation concludes that the missile will not puncture the cask water jacket.

The analysis of the effects of the wind-driven missiles and the bounding evaluation of the effects of the DBT (Section D3.4.2.6) show that calculated energies for the design basis wind are well bounded by the cask drop analyses. These analyzed energies are found to be acceptable in Section D3.4.2.1, which concludes that the cask system is able to withstand any induced stresses from handling/drop events without compromising the confinement function of the LWR canister.

Fire – The DBF is analyzed in Chapter D3.0 and summarized in Section D3.4.2.7. The analysis determines the DBF to be bounded by the exposure of the NAC-1 cask and associated ISO shipping container to the 10 CFR 71.73(c)(3) transportation fire.

The analysis of the NAC-1 cask subjected to the transportation fire scenario of 10 CFR 71.73(c)(3) is provided in *Safety Analysis Report for Nuclear Fuel Services, Inc., Spent Fuel Shipping Cask Model No. NFS-4* (NFS 1972) for an internal thermal load of 11.5 kW, and in the NAC-1 SAR (NAC-E-804) for an internal 750 W thermal load. These analyses both analyze the NAC-1 cask with and without the ISO shipping container and provide bounding conditions for the fire accident scenario for use of the NAC-1 shipping cask for storage of the commercial LWR fuel. The LWR fuel has a maximum internal thermal load of 405 W. The transportation SAR and the NAC-1 SAR analyses are therefore considered sufficient to demonstrate the integrity of the NAC-1 cask for the fire accident conditions. The cask integrity is preserved under both of these conditions for storage of the 300 Area LWR fuel.

The analysis in Section D3.4.2.7 concludes that the NAC-1 cask and the LWR canister will withstand any stresses or temperatures induced as a result of fire, and that the canister structural integrity required to ensure that an uncontrolled release of radioactive material will not occur is not compromised and the canister remains leaktight.

D4.4.5.5 NAC-1 Cask System Controls (Technical Safety Requirements).

The assumptions associated with the NAC-1 cask storage system that require TSRs to ensure performance of their safety function of confinement control are as follows:

- The lift height of 30 ft is not exceeded for the ISO container.
- The crane hook and block lift height of 56 ft is not exceeded.
- The ISO hydraulic spreader bar lift height of 6 ft above the ISO container shall not be exceeded.
- The number of individual fuel rods in an NAC-1 cask does not exceed a maximum of 179 PWR rods, or 96.5 BWR rods consolidated with 17 PWR rods.

- The NAC-1 cask in the ISO container shall be handled with an approved crane. Only the Manitowoc 4000 150T crane with a Model 22 80-ft boom and the Manitowoc 4100 250T crane with a Model 27 80-ft boom were analyzed in this Final Safety Analysis Report. A program shall be in place to ensure that only approved cranes are used.
- The NAC-1 cask shall be placed on a concrete pad.

D4.5 REFERENCES

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10 CFR 830, "Nuclear Safety Management," *Code of Federal Regulations*, as amended.

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CHAPTER D5.0

DERIVATION OF TECHNICAL SAFETY REQUIREMENTS

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CONTENTS

LIST OF TABLES

LIST OF TERMS

AC	Administrative Control
ALARA	as low as reasonably achievable
BED	building emergency director
BWR	boiling water reactor
CCC	core component container
DFA	driver fuel assembly
DOE	U.S. Department of Energy
FFTF	Fast Flux Test Facility
FSAR	final safety analysis report
ISA	interim storage area
ISC	interim storage cask
ISO	International Standards Organization
LCO	Limiting Condition for Operation
LWR	light water reactor
NPH	natural phenomena hazard
NRF	Neutron Radiography Facility
PWR	pressurized water reactor
SNF	spent nuclear fuel
SSC	structure, system, and component
TSR	Technical Safety Requirement

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D5.0 DERIVATION OF TECHNICAL SAFETY REQUIREMENTS

D5.1 INTRODUCTION

A description of the essential features of the Spent Nuclear Fuel (SNF) Project derivation of Technical Safety Requirements (TSRs) is provided in Section 5.1 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1.

D5.2 REQUIREMENTS

The requirements that form the basis for the SNF Project derivation of TSRs are identified in Section 5.2 of HNF-3553, Volume 1.

D5.3 TECHNICAL SAFETY REQUIREMENTS COVERAGE

The 200 Area Interim Storage Area (ISA) TSRs for the analyzed hazards and accidents are summarized in Table D5-1. This table lists TSR controls in accordance with Chapter D4.0 and the accident analyses in Chapter D3.0. Table D5-1 provides a cross-reference of the respective accident analysis section to the relevant subsections within Section D5.5, where derivation details are arranged by TSR control.

The necessary and sufficient TSR controls are established based on consideration for public safety, significant defense in depth, significant worker safety, and for maintaining radiological consequences below risk evaluation guidelines. HNF-3553, Volume 1, Section 5.3 contains details applicable to all SNF Project facilities. Section D5.3.2 contains information specific to the 200 Area ISA, in addition to that provided in Section 5.3.2 of HNF-3553, Volume 1.

D5.3.1 Criteria

The control selection criteria used for the SNF Project are described in Section 5.3.1 of HNF-3553, Volume 1.

Table D5-1. Final Safety Analysis Report Section and Technical Safety Requirement Cross Reference. (3 sheets)

Final Safety Analysis Report section	Technical Safety Requirement	Control basis	
D3.4.2.4.5	There shall be no more than seven fuel assemblies in a CCC.	AC5.8	Source Inventory Receipt Acceptance
D4.3.1.5	Each CCC shall contain six DFAs in the outer positions, seven DFAs including the center position, or six Ident-69 containers in the outer positions. The loaded CCC can contain a mixture of fuel assemblies and pin containers such that the total is either six or seven, and the number of pin containers is five or less. No partially filled CCCs are permitted.	AC5.8	Source Inventory Receipt Acceptance
D4.3.1.5	There shall be no more than a total of 1,519 pins in a CCC container to protect the fuel rod rupture pressure calculation assumptions.	AC5.8	Source Inventory Receipt Acceptance
D6.3.1.1	Only authorized fuel, as identified in Chapters D2.0 and D6.0, can be received at the ISA.	AC5.8	Source Inventory Receipt Acceptance
D4.3.1.5	Only intact fuel shall be placed into a CCC.	AC5.8	Source Inventory Receipt Acceptance
D3.4.2.4.5 D4.3.2.5 D4.4.5.5	The number of individual fuel rods in the NAC-1[a] cask does not exceed a maximum of 179 PWR rods or 96.5 BWR rods consolidated with 17 PWR rods.	AC5.8	Source Inventory Receipt Acceptance
D3.4.2.1.5 D4.4.1.5	The ISC cannot be lifted more than 8 ft above the ground or storage pad.	AC5.9	Lift Restrictions
D3.4.2.1.5 D4.4.1.5	The ISC cannot be lifted over an object that is taller than 4 ft or containing radioactive materials.	AC5.9	Lift Restrictions
D4.4.1.5	Crane loads other than the ISC rigging shall not be operated over a loaded ISC.	AC5.9	Lift Restrictions
D4.4.2.5	The Rad-Vault[b] can only be lifted empty with the lid removed.	AC5.9	Lift Restrictions

Table D5-1. Final Safety Analysis Report Section and Technical Safety Requirement Cross Reference. (3 sheets)

Final Safety Analysis Report section	Technical Safety Requirement	Control basis	
D3.4.2.1.5 D4.4.2.5	The Rad-Vault[b] lid shall not be handled at a height greater than 12 in. above the top of the Rad-Vault.[b]	AC5.9	Lift Restrictions
D4.4.2.5	The Rad-Vault[b] lid shall be replaced after each activity within the Rad-Vault.[b]	AC5.9	Lift Restrictions
D4.4.2.5	Crane loads other than NRF TRIGA[c] casks, DOT-6M[d] containers, rigging, and the Rad-Vault[b] lid shall not be handled over a loaded Rad-Vault.[b]	AC5.9	Lift Restrictions
D3.4.2.1.5 D4.4.3.5 D4.4.4.5	The NRF TRIGA[c] casks and DOT-6M[d] containers cannot be lifted more than 109 in. (approximately 9.1 ft) above the ground or more than 21 in. above the top of the Rad-Vault[b] with the lid removed.	AC5.9	Lift Restrictions
D3.4.2.1.5	The NAC-1[a] package cannot be lifted over other objects except the transport trailer.	AC5.9	Lift Restrictions
D3.4.2.1.5 D4.4.5.5	The NAC-1[a] package cannot be lifted more than 30-ft from the surface of the 200 Area ISA storage pad.	AC5.9	Lift Restrictions
D3.4.2.1.5	The ISO hydraulic spreader bar lift height of 6 ft above the ISO container shall not be exceeded.	AC5.9	Lift Restrictions
D4.4.5.5	The crane hook and block lift height of 56 ft is not exceeded.	AC5.9	Lift Restrictions
D3.4.2.5.5 D3.4.2.6.5	The ISC shall be placed on a concrete pad.	AC5.10	Spacing and Placement
D3.4.2.4.5 D4.4.1.5	The ISC minimum spacing array area shall be no less than 24 in. by 44 in. between ISCs measured edge to edge.	AC5.10	Spacing and Placement
D3.4.2.4.5 D4.4.1.5	A method shall be established for indicating proper ISC placement (e.g., painting an area on the storage pad).	AC5.10	Spacing and Placement
D4.4.1.5	After unloading an ISC from the transport vehicle, verify the ISC is properly positioned on the storage pad and the minimum spacing requirement is met.	AC5.10	Spacing and Placement

Table D5-1. Final Safety Analysis Report Section and Technical Safety Requirement Cross Reference. (3 sheets)

Final Safety Analysis Report section	Technical Safety Requirement	Control basis
D3.4.2.5.5 D3.4.2.6.5 D4.4.2.5	The Rad-Vault[b] shall be placed on compacted gravel.	AC5.10 Spacing and Placement
D3.4.2.5.5 D3.4.2.6.5 D4.4.5.5	The NAC-1[a] cask shall be placed on a concrete pad.	AC5.10 Spacing and Placement
D3.4.2.2.5 D4.4.1.5 D4.4.5.5	The ISC and NAC-1[a] cask lifts are restricted to the use of an approved crane. The Manitowoc 4000 150T crane with Model 22 80-ft boom and Manitowoc 4100 250T crane with Model 27 80-ft boom have been analyzed and are approved. A program shall be in place to ensure that only approved cranes are used.	AC5.11 Crane Utilization
D3.4.2.2.5 D4.4.1.5 D4.4.5.5	All lifts are to be considered critical lifts, as discussed in the Hanford Site hoisting and rigging criteria.	AC5.11 Crane Utilization
D4.4.2.5 D4.4.3.5 D4.4.4.5	The NRF TRIGA[c] cask, DOT-6M[d] container, and Rad-Vault[b] lid shall be handled with an approved crane. The Hanford Site hoisting and rigging criteria (DOE/RL-92-36)[e] for a critical lift shall be imposed. A program shall be in place to ensure that only approved cranes are used.	AC5.11 Crane Utilization
D3.4.2.7.5 D4.4.1.5	Fire loadings are to be controlled per the fire hazard analysis.	AC5.12 Combustible Loading Limits

[a] NAC-1 casks are manufactured by Nuclear Assurance Corporation.
[b] Rad-Vault is a trademark of Chem-Nuclear Systems, Incorporated.
[c] TRIGA is a trademark of General Dynamics Corporation.
[d] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.
[e] DOE/RL-92-36, 1993, *Hanford Site Hoisting and Rigging Manual*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

AC = administrative control.
BWR = boiling water reactor.
CCC = core component container.
DFA = driver fuel assembly.
ISA = interim storage area.
ISC = interim storage cask.
ISO = International Standards Organization.
NRF = Neutron Radiography Facility.
PWR = pressurized water reactor.

D5.3.2 Safety Structures, Systems, and Components Not Provided with Technical Safety Requirement Coverage

All safety-class and safety-significant structures, systems, and components (SSCs) have been provided with TSR coverage, as identified in this chapter. All safety SSCs are provided with TSR coverage with an Administrative Control (AC) program. There are no Limiting Conditions for Operation (LCOs) identified for the 200 Area ISA.

D5.4 DERIVATION OF FACILITY MODES

D5.4.1 Operational Modes

Modes are used to specify when safety limits, LCOs, or surveillance requirements are required. 200 Area ISA operations include the movement and storage of the casks and inspections and surveillances. The casks and canisters are the only barriers to the uncontrolled release of radioactive material. The safety-class SSCs are passive and protect against criticality.

The normal operational mode for the 200 Area ISA (after movement and storage of the casks) is long-term interim storage, which is the "Operation" mode. There are no clear operational distinctions to provide more than one mode for safety-class SSCs. The hazard presented by the spent fuel in the casks/containers cannot be shut off or reduced by operational changes. Therefore, the 200 Area ISA will have only one mode, which will be designated as the "Operation" mode. The 200 Area ISA is capable of receiving or transferring casks between locations within the facility during this mode. Routine operational and maintenance activities will be performed (e.g., surveillances, inspections, radiological monitoring, and repairs).

D5.4.2 Minimum Staffing Levels

During long-term interim storage, there is no manned shift requirement. Assignment of radiological and maintenance personnel on a periodic basis is coordinated with the Canister Storage Building and is considered adequate to perform the inspections, surveillance, repairs, and monitoring necessary to protect the health and safety of the public and the collocated worker.

Qualification training for the assigned personnel is addressed in Chapter D12.0. Emergency response is addressed in Chapter D15.0.

Normal long-term interim storage operations. Determination of the unmanned shifts during normal interim storage operations assumes that routine inspection and surveillance are being conducted by Canister Storage Building personnel. An inspection or surveillance is considered a low-difficulty task.

Abnormal conditions. Manning during abnormal conditions is the same as for normal/long-term interim storage conditions.

Emergency conditions. The assigned personnel during emergency conditions are needed to ensure an appropriate response to the spectrum of accidents analyzed in Chapter D3.0 (hazardous and non-hazardous). The minimum assigned personnel must make prompt initial notifications and implement initial protective actions to preclude or reduce the exposure of individuals affected by the hazards or unsafe conditions during an emergency.

The Canister Storage Building facility manager or designee is the building emergency director (BED) with the primary responsibility for assessing the event and implementing protective actions at the 200 Area ISA. The BED also makes onsite notifications, implements emergency management procedures, implements facility emergency plans, classifies events, and controls event response. The BED requests support services, as necessary, to perform administrative functions and the minimum functions required to ensure the health and safety of the public, onsite workers, and the environment.

D5.5 TECHNICAL SAFETY REQUIREMENT DERIVATION

Controls associated with the 200 Area ISA are derived in Table D5-1 and are discussed in this section, except for AC 5.1 through AC 5.6, which will be included in the 200 Area ISA TSR document. AC 5.1, "Purpose," AC 5.2, "Contractor Responsibility," and AC 5.3, "Compliance," define the need for subsequent AC provisions and clarify responsibility according to good management practice. AC 5.4, "Technical Safety Requirement Violation," AC 5.5, "Occurrence Reporting," and AC 5.6, "Organization," satisfy the requirements of U.S. Department of Energy (DOE) Order 5480.22, *Technical Safety Requirements*, Section 9.e.(5). This DOE Order requires ACs for reporting deviations and identifying staffing requirements. Worker safety program requirements are addressed in other Final Safety Analysis Report (FSAR) sections based on design and operations information.

D5.5.1 Limiting Conditions For Operation

During long-term storage operation, there are no identified LCOs at the 200 Area ISA.

D5.5.2 Safety Limits

During long-term storage, there are no identified Safety Limits that could be exceeded at the 200 Area ISA.

D5.5.3 Administrative Controls

AC 5.1 through AC 5.6, in accordance with DOE Order 5480.22, are identified above. Additional controls are defined in the subsections that follow.

D5.5.3.1 AC5.7 – Nuclear Criticality Safety.

Purpose. This control protects the assumptions of the nuclear criticality evaluation in Chapter D6.0 and the accident analyses in Chapter D3.0 to ensure that operations and storage at the 200 Area ISA prevent inadvertent nuclear criticality. The purpose of this AC is to protect features that are relied on to preclude a criticality event at the 200 Area ISA. Key elements of this programmatic AC are derived from contractor procedures that include requirements for criticality safety evaluations, criticality prevention specifications, and criticality training. The criticality controls identified in Chapter D6.0 are as follows:

- Obtain the source inventory from the shipping organization to verify the containers meet the TSR for geometry control. Storage at the 200 Area ISA ensures that geometry is maintained. Water intrusion into the Fast Flux Test Facility (FFTF) core component container (CCC) and the light water reactor (LWR) canister is prevented by the individual system designs.
- Incorporate lift restrictions as required by AC 5.9.

Derivation Criterion. A nuclear criticality program is required as one of the standard ACs, according to DOE Order 5480.22, Section 9.e.(5).

D5.5.3.2 AC5.8 – Source Inventory Receipt Acceptance.

Purpose. This AC ensures that shipments received at the ISA only include hazardous and radiological materials that have been analyzed in this FSAR Annex. The key elements of this AC are as follows:

- Acceptance criteria are established such that fuel type, quantity, and configuration are limited to those authorized in this FSAR Annex. The acceptance criteria shall include the following:
  - There shall be no more than seven fuel assemblies in a CCC.
  - Each CCC shall contain six driver fuel assemblies (DFAs) in the outer positions, seven DFAs including the center position, or six Ident-69 containers in the outer positions. The loaded CCC can contain a mixture of fuel assemblies and pin containers such that the total is either six or seven, and the number of pin containers is five or less. No partially filled CCCs are permitted.
  - There shall be no more than a total of 1,519 pins in a CCC container to protect the fuel rod rupture pressure calculation assumptions.
  - Only authorized fuel, as identified in Chapters D2.0 and D6.0, can be received at the ISA.
  - Only intact fuel shall be placed into a CCC.
  - The number of individual fuel rods in the NAC-1 cask does not exceed a maximum of 179 pressurized water reactor (PWR) rods, or 96.5 boiling water reactor (BWR) rods (95 full-length rods plus 6 quarter-length sealed segments) consolidated with 17 PWR rods.
- Procedures are established to ensure that shipments meet acceptance criteria.
- Records are kept, maintained, and available for review that document that the material stored at the ISA meets the acceptance criteria.

Derivation Criteria. These controls are necessary to ensure that the ISA does not accept material that has not been analyzed in the design basis.

D5.5.3.3 AC5.9 – Lift Restrictions.

Purpose. This AC focuses on the importance of the planned operational steps that were identified as critical assumptions in the accident analyses in Chapter D3.0 and delineated in Chapter D4.0. The key elements of this AC, which preclude criticality or loss of confinement accidents, are as follows:

- The interim storage cask (ISC) cannot be lifted more than 8 ft above the ground or storage pad.
- The ISC cannot be lifted over an object that is taller than 4 ft or that contains radioactive materials.
- Crane loads other than the ISC rigging shall not be operated over a loaded ISC.
- The Rad-Vault can only be lifted empty with the lid removed.
- The Rad-Vault lid shall not be handled at a height greater than 12 in. above the top of the Rad-Vault.
- The Rad-Vault lid shall be replaced after each activity within the Rad-Vault.
- Crane loads other than Neutron Radiography Facility (NRF) TRIGA casks, DOT-6M containers, rigging, and the Rad-Vault lid shall not be handled over a loaded Rad-Vault.
- The NRF TRIGA casks and DOT-6M containers cannot be lifted more than 109 in. (approximately 9.1 ft) above the ground or more than 21 in. above the top of the Rad-Vault with the lid removed.
- The NAC-1 package cannot be lifted over other objects except the transport trailer.
- The NAC-1 package cannot be lifted more than 30-ft from the surface of the 200 Area ISA storage pad.
- The crane hook and block lift height of 56 ft is not exceeded.
- The International Standards Organization (ISO) hydraulic spreader bar lift height of 6 ft above the ISO container shall not be exceeded.
- All lifts are to be considered critical lifts, as discussed in the hoisting and rigging criteria in DOE/RL-92-36, *Hanford Site Hoisting and Rigging Manual*.

Derivation Criteria. These controls are necessary to support the assumptions of the safety analysis. The assumptions rely on personnel actions to ensure that they remain valid.

D5.5.3.4 AC5.10 – Spacing and Placement.

Purpose. This AC focuses on the importance of the critical assumptions identified in the accident analyses in Chapter D3.0 and delineated in Chapter D4.0. The key elements of this AC, which preclude loss of confinement accidents, are as follows:

- The ISC shall be placed on a concrete pad.
- The ISC minimum spacing array area shall be no less than 24 in. by 44 in. between ISCs measured edge to edge.
- A method shall be established for indicating proper ISC placement (e.g., painting an area on the storage pad).
- After unloading an ISC from the transport vehicle, verify the ISC is properly positioned on the storage pad and the minimum spacing requirement is met.
- The Rad-Vault shall be placed on compacted gravel.
- The NAC-1 cask shall be placed on a concrete pad.

Derivation Criteria. These controls are necessary to support the assumptions of the safety analysis.

D5.5.3.5 AC5.11 – Crane Utilization.

Purpose. This AC ensures that only cranes and booms that are within the calculational analysis are used. The key elements of this AC, which preclude loss of confinement accidents, are as follows:

- The ISC and NAC-1 cask lifts are restricted to the use of an approved crane. The Manitowoc 4000 150T crane with Model 22 80-ft boom and the Manitowoc 4100 250T crane with Model 27 80-ft boom have been analyzed and are approved. A program shall be in place to ensure that only approved cranes are used.
- The NRF TRIGA cask and DOT-6M container shall be handled with an approved crane. The Hanford Site hoisting and rigging criteria (DOE/RL-92-36) for a critical lift shall be imposed. A program shall be in place to ensure that only approved cranes are used.
- All lifts are to be considered critical lifts, as discussed in the Hanford Site hoisting and rigging criteria (DOE/RL-92-36).

Derivation Criteria. These controls are necessary to support the assumptions of the safety analysis.

D5.5.3.6 AC5.12 – Combustible Loading Limits.

Purpose. This AC provides protection against exceeding the combustible loadings assumed in the fire hazard analysis. The key elements of this AC, which preclude loss of confinement accidents, are as follows:

- Fire loadings are to be controlled in accordance with the fire hazard analysis.

Derivation Criteria. These controls are necessary to support the assumptions of the fire hazard analysis.

D5.6 DESIGN FEATURES

Design features for the 200 Area ISA that, if altered or modified, would have a significant effect on safety are listed below. Descriptions of these design features are provided in Chapter D2.0; the safety functions they perform are described in Chapter D4.0.

D5.6.1 Fast Flux Test Facility Spent Nuclear Fuel Core Component Container

Design features for the FFTF CCC include the following:

- Criticality geometry control
- Corrosion-resistant spent fuel storage
- SNF protection against all credible drops
- Radiological shielding for ALARA (as low as reasonably achievable) protection
- The CCC Helicoflex seal shall have a design temperature of at least 500 °F.

D5.6.2 Light Water Reactor Canister

Design features for the LWR canister include the following:

- Criticality geometry control
- SNF confinement
- Leaktight LWR canister to prevent water intrusion
- Shielding overpack for weather and natural phenomena hazards (NPH) protection
- The individual loose rods must be retained within the 8-in. square rod consolidation container.

D5.6.3 Fast Flux Test Facility Interim Storage Cask

Design features for the FFTF ISC include the following:

- SNF secondary confinement
- Passive heat removal

- CCC environmental protection
- Radiological shielding for ALARA protection
- SNF drop protection
- Weather protection with the ISC cover installed.

D5.6.4 Neutron Radiography Facility TRIGA Cask

Design features for the NRF TRIGA cask include the following:

- Confinement and shielding for the TRIGA fuel
- 20-year storage life requirement of the cask metallic seals
- Seals must meet the leak-tight requirement for transportation.

D5.6.5 DOT-6M and 2R Containers

Design features for the DOT-6M package with the inner 2R container include the following:

- SNF confinement
- Radiological shielding for ALARA protection.

D5.6.6 Rad-Vault

Design features for the Rad-Vault include the following:

- Supplemental shielding for NRF TRIGA casks and DOT-6M containers
- Heat shield for fires
- NPH protection for wind, wind missiles, tornadoes, and volcanic ashfall
- Weather cover to mitigate snow, dust, sandstorm, and rain.

D5.6.7 NAC-1 Cask and International Standards Organization Shipping Container

Design features for the NAC-1 cask include the following:

- Radiological shielding for ALARA protection
- Protection for the LWR canister
- Protection from stresses from natural phenomena and accident events.

D5.6.8 At-Grade Storage Pads and Compacted Gravel Structure at 200 Area Interim Storage Area

Design features for grading include the following:

- Maintain site grading such that the probable maximum precipitation event does not allow water to degrade the storage pad (undercut concrete pad).
- Maintain site grading such that the accumulation of runoff water cannot result in a pad overflow of water.

D5.7 INTERFACES WITH TECHNICAL SAFETY REQUIREMENTS FROM OTHER FACILITIES

There are no TSRs of other SNF Project facilities that affect the 200 Area ISA. However, there is interaction of the ISA with existing Canister Storage Building facilities for surveillance activities and central alarm notification.

D5.8 REFERENCES

DOE Order 5480.22, 1996, *Technical Safety Requirements*, U.S. Department of Energy, Washington, D.C.

DOE/RL-92-36, 1993, *Hanford Site Hoisting and Rigging Manual*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

HNF-3553, 2000, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Vol. 1, Rev. 0-B, Fluor Hanford, Incorporated, Richland, Washington.

CHAPTER D6.0

PREVENTION OF INADVERTENT CRITICALITY

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CONTENTS

LIST OF TABLES

LIST OF TERMS

AL	aluminum
BWR	boiling water reactor
CCC	core component container
CSER	criticality safety evaluation report
DBA	design basis accident
DFA	driver fuel assemblies
DOE	U.S. Department of Energy
FFCR	fuel follower control rod
FFTF	Fast Flux Test Facility
ISA	interim storage area
ISC	interim storage cask
ISO	International Standards Organization
LWR	light water reactor
MOX	mixed oxide
NRC	U.S. Nuclear Regulatory Commission
NRF	Neutron Radiography Facility
PWR	pressurized water reactor
SNF	spent nuclear fuel
SS	stainless steel

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D6.0 PREVENTION OF INADVERTENT CRITICALITY

D6.1 INTRODUCTION

This chapter provides an evaluation of the potential for a criticality accident in the 200 Area Interim Storage Area (ISA). Controls required to prevent criticality in the ISA are also discussed. Chapter 6.0 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1, provides an overall description of the Spent Nuclear Fuel (SNF) Project criticality prevention program.

The scope of the criticality analysis and control strategy for the ISA includes criticality hazards associated with the handling and storage of SNF at the ISA. Specifically, this includes receipt and unloading of sealed casks from transportation vehicles, movement of casks within the ISA facility, and storage of casks in the facility for a period of up to 40 years. Activities excluded from the scope are: (1) transport of the SNF to the ISA facility, and (2) handling and loading of SNF into casks at the Fast Flux Test Facility (FFTF) and the 300 Area prior to transport to the ISA. However, the impact of abnormal events or accidents are considered that may affect the condition or configuration of casks delivered to the ISA (e.g., the criticality hazard associated with receiving a misloaded cask is analyzed). The criticality safety of transportation of SNF to the ISA facility is analyzed in WHC-SD-TP-SARP-008, *Safety Analysis Report for Packaging NRF TRIGA Packaging*, and WHC-SD-TP-SARP-010, *Safety Analysis Report for Packaging (Onsite) Interim Storage Cask.*

D6.2 REQUIREMENTS

The requirements that form the basis of criticality prevention are identified in Section 6.2 of HNF-3553, Volume 1.

This chapter also addresses implementation of requirements to achieve U.S. Nuclear Regulatory Commission (NRC) equivalency in the design of the ISA with respect to criticality prevention based on the double-contingency principle. The double-contingency principle is stated in U.S. Department of Energy (DOE) Order 5480.24, *Nuclear Criticality Safety*, as follows: "Process designs shall incorporate sufficient factors of safety to require at least two unlikely, independent and concurrent changes in process conditions before a criticality accident is possible." In HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements*, the SNF project established a k_{eff} limit of 0.95 (including allowance for uncertainty) for the ISA. As applied in this chapter, contingencies include failure of an engineered feature incorporated in the ISA design or failure of an administrative control affecting criticality safety at the ISA.

D6.3 CRITICALITY CONCERNS

D6.3.1 Fissile Material Form and Inventory

As described in Chapter D2.0, four different dry cask storage containers will be stored in the ISA, including the following:

- Interim Storage Casks (ISC) containing FFTF fuel or experimental targets.
- Six Neutron Radiography Facility (NRF) TRIGA[1] casks containing TRIGA fuel.
- Two DOT-6M[2] casks containing the TRIGA fuel follower control rods (FFCRs).
- NAC-1[3] casks containing 300 Area light water reactor (LWR) fuel.

The layout of casks on storage pads and within the Rad-Vault[4] at the ISA facility is described in Section D2.3 and illustrated in Figure D2-1. The design and specifications of the casks themselves are described in Section D2.6. Details of the fuel design and inventory stored in the casks are found in Section D2.5. For the purposes of criticality safety, the fissile material form and contents of the fuel stored in the casks are summarized in the subsections that follow.

D6.3.1.1 Fast Flux Test Facility Fuel.

Several different types of fuel pins were analyzed in the criticality safety evaluation reports (CSERs) prepared for the storage of FFTF fuel (see Section D2.5.1.1 for details), as follows:

- Mixed oxide (MOX) fuel pins – The MOX fuel pins contain uranium dioxide and plutonium dioxide in pellet form clad with stainless steel. Slight variations in the enrichment of the fuel exist within the inventory. However, enrichment is bounded by a maximum 29.28 wt% ^{239}Pu and ^{241}Pu and a minimum 11.63 wt% ^{240}Pu.
- Metal fuel pins – The metal fuel pins contain a sodium-bonded uranium/zirconium alloy clad with stainless steel. The metal fuel is of two enrichments: 31 wt% ^{235}U or 32.4 wt% ^{235}U. Note: Storage of metal fuel is not currently authorized at the ISA (see Section D2.5.1.1), but is included in this chapter for completeness since a criticality analysis has already been performed for the metal fuel.
- Certain experimental fuel pins and repackaged fuel debris – This fuel is classified as equivalent to MOX fuel (see Section D2.5.1.1 for details). Note: Storage of experimental fuel debris is not currently authorized at the ISA (see Section D2.5.1.1), but is included in this chapter for completeness since a criticality analysis has already been performed for the fuel debris.

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[1] TRIGA is a trademark of General Dynamics Corporation.

[2] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.

[3] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

[4] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.

Note: MOX and metal fuel were analyzed for criticality safety at the ISA. Other FFTF fuel types (e.g., those contained in certain "test fuel assemblies") (see Section D2.5.1.1) cannot be accepted at the 200 Area ISA without further criticality analysis.

The FFTF CSERs analyzed core component container (CCC) tubes loaded with a combination of the following: (Note: storage of metal fuel and experimental fuel debris are not yet authorized for the ISA. However, storage configurations of these fuels that are acceptable from a criticality safety standpoint are discussed here since they have already been analyzed in the FFTF CSERs.)

- Driver fuel assemblies (DFAs) consisting of a hexagonal array of 217 MOX fuel pins.
- Ident-69 containers with varying numbers of MOX fuel pins (up to 217).
- MOX fuel test assemblies consisting of a hexagonal array of 217 or 169 fuel pins.
- Metal fuel test assemblies consisting of a hexagonal array of 169 metal fuel pins.
- Modified Ident-69s containing returned experimental fuel pins and repacked fuel pin debris.

FFTF fuel will be stored in a given ISC subject to the following restrictions:

- No Ident-69, metal assemblies, or modified Ident-69s with repackaged fuel are allowed in the center CCC position. (Note: Ident-69 containers are too long to fit in the center position without modification.)
- The center tube of the CCC must be empty if there are six Ident-69s.
- Combinations of metal assemblies and Ident-69s are not allowed.
- Modified Ident-69s must not contain over 4 kg of MOX fuel and the enrichment must be <31 wt% plutonium.

These restrictions were developed in the FFTF CSERs to ensure that safe subcritical limits are not violated during handling and loading of the ISCs at FFTF.

Numerous possibilities exist for the actual loading of the six CCC tubes with the above tube loading options. The k_{eff} of various bounding configurations for the CCC tube is analyzed in Section D6.3.2.

The fissile material inventory for each fuel type and storage configuration within a CCC storage tube is shown in Table D6-1. This inventory represents the upper bound on plutonium and uranium enrichment.

Table D6-1. Fuel Inventories.

Isotope	Core Component Container Tube Configuration (mass in grams)			
	Mixed oxide fuel in driver fuel assemblies and mixed oxide test assemblies	Metal fuel assembly	Mixed oxide fuel in Ident-69 pin container	Experimental fuel in modified Ident-69 pin container
^{239}Pu	8,421		9,042	953
^{240}Pu	1,152		1,237	130
^{241}Pu	100		108	11
^{242}Pu	19		20	2
^{235}U	47	13,971	50	5
^{238}U	23,274	29,102	24,990	2,634
^{241}Am	42		45	5

D6.3.1.2 Neutron Radiography Facility TRIGA Fuel.

As described in Section D2.5.1.2, two types of fuel elements will be stored in the six NRF TRIGA casks: (1) stainless steel-clad fuel elements with ceramic zirconium hydride/uranium fuel having 8.5 wt% uranium content with 20% enrichment, and (2) aluminum-clad fuel elements of a similar specification. Two FFCRs will also be stored in the Rad-Vault in DOT-6M casks. The uranium content (shown in Table D2-8) varies slightly by fuel element, but does not exceed 41 g of ^{235}U.

There are a total of 66 aluminum-clad fuel elements, 33 stainless steel-clad fuel elements, and 2 FFCR elements. The total fissile material content of the Rad-Vault will not exceed 4 kg of ^{235}U.

D6.3.1.3 Commercial Light Water Reactor Fuel.

The fuel stored in the NAC-1 casks is described in detail in Section D2.5.3. Six casks will be used to store five pressurized water reactor (PWR) assemblies and a special loose pin container with pins from two boiling water reactor (BWR) assemblies, and six Segmented Rod Program segmented rods from two commercial power reactors. The fuel rods consist primarily of uranium oxide fuel clad in zircaloy, with ^{235}U enrichment varying from 2.45 to 3.6 wt%. From Table D2-11, the total ^{235}U content of all the fuel stored in the NAC-1 casks is calculated to be less than 63 kg.

D6.3.1.4 Total Interim Storage Area Inventory.

The total ISA inventory consists of fissionable material found in 210 FFTF DFAs, 65 FFTF test DFAs, and 11 FFTF test fuel assemblies (those classified as equivalent to MOX fuel), 5 LWR fuel assemblies, 2 FFCRs, 118 loose LWR pins, and 101 TRIGA fuel pins. The total FFTF fuel inventory was inferred from the total plutonium and uranium mass in FFTF fuel listed in Section D2.5.1.1. The fissionable material totals in Table D6-2 are upper bounds on plutonium and uranium enrichment used in the criticality analysis and may differ slightly from actual inventories due to variations in enrichment.

Table D6-2. Total Interim Storage Area Inventories.

Isotope	Interim Storage Area Component (mass in kg)			
	Two interim storage cask pads	NAC-1 pad[a]	Rad-Vault	Total (kg)
^{239}Pu	2,085			2,085
^{240}Pu	285			285
^{241}Pu	24			24
^{242}Pu	4.7			4.7
^{235}U	95	63	4	162
^{238}U	5,937	2,230	20	8,187
^{241}Am	10			10

[a] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

D6.3.2 Criticality Hazards

Table D6-3 identifies criticality hazards associated with handling and storage of the SNF. These hazards apply to all four types of casks, with exceptions as noted.

Changes in k_{eff} of the casks or groups of casks due to the above hazards are analyzed in Section D6.3.3. The k_{eff} analysis includes the effect of multiple hazardous conditions from Table D6-3 occurring at the same time. For example, the effect of flooding of a cask in which fuel has disintegrated due to a corrosion phenomenon called hot-cell rot is considered for the ISCs.

D6.3.3 Analysis of Hazardous Conditions

Evaluations of neutron multiplication factors (k_{eff}) for hazardous conditions affecting SNF stored at the ISA have been performed in several CSERs that are summarized in this section. These documents were prepared several years prior to development of the SNF Project safety analysis report. The existing CSERs formed the authorization basis for transportation and storage of SNF at the FFTF 400 Area ISA. These historical CSERs employed several different methods for calculating bias and uncertainty in k_{eff} and determining the upper bound on k_{eff}, which are summarized in this section. For comparison to the safe subcritical limit of 0.95, the uncertainty and bias (at the 95% confidence level) reported by the CSERs have been added to the values of k_{eff} calculated in the CSERs, as described in Section 6.3 of HNF-3553, Volume 1. These "upper bound" values of k_{eff} have been reported in the discussion of analysis results below (as opposed to the "unadjusted" value of k_{eff} calculated by the analysis models in the CSERs).

Table D6-3. Criticality Hazards Evaluation. (2 sheets)

Event Label	Description	Cause	Hazardous effect	Probability	k_{eff} analyzed?
Interim storage area flooding	Large scale flooding in the ISA	Natural phenomenon such as rain or snow melt	Water introduction into cask(s) increases k_{eff} in cask(s)	Not credible, ISA site is above the flood plain.	Yes, can be inferred from the effects of single cask flooding.
Cask flooding	Localized flooding into one or several casks	Fire hydrant, rain, snow melt, fire suppression	Water introduction into cask(s) increases k_{eff} in cask(s)	Not credible for ISCs or LWR canister. Very unlikely for TRIGA[a] fuel, due to multiple barriers against water introduction.[b]	Yes
Hot-cell rot	Fuel pins and/or assembly ducts disintegrate over time	Stress corrosion cracking under adverse conditions within cask	Fuel relocated to a more reactive geometry	Anticipated for FFTF fuel, corrosion effects not well known so fuel pin and assembly duct failure is assumed. Not credible for TRIGA[a] and LWR fuel.[c]	Yes
Tipping/ dropping	Cask dropped or crushed during handling	Crane or transport vehicle failure, operational error while handling casks	Fuel relocated to a more reactive geometry	Unlikely[d]	Yes
Neutron interaction between casks	Casks are arranged too close on a pad or there are too many casks	Administrative error in cask placement	Increased k_{eff} due to neutron interaction between casks	Unlikely	Yes, infinite, close packed cask arrays were analyzed. The thickness of the cask structural materials minimizes any interaction.
Seismic	DBA seismic event tips cask(s) over	N/A	Cask tips. Fuel relocated to a more reactive geometry	Not credible, tipping of cask shown to be not credible by seismic analysis.	Yes
Fire	Fire within the ISA perimeter	Range fire, brush accumulation within ISA perimeter, vehicle fuel spill	Increased temperature changes k_{eff} of casks	Not credible. Worst-case fire has been analyzed and shown not to have a significant effect on fuel temperature (Section D3.4.2.7).	No

Table D6-3. Criticality Hazards Evaluation. (2 sheets)

Event Label	Description	Cause	Hazardous effect	Probability	k_{eff} analyzed?
Cask misloading	Cask is loaded with incorrect fuel assembly(s)	Administrative error when loading casks at the facility of origin	Increased k_{eff} due to incorrect fuel arrangement	Unlikely, due to administrative controls on cask loading before transfer to the ISA.[e]	Yes

[a] TRIGA is a trademark of General Atomics.

[b] Water intrusion into an ISC was judged to be not credible due to the multiple barriers that exist to prevent this hazard. For example, the ISC has an environmental cover to prevent snow and rain intrusion, and two metallic O-ring seals that must fail to allow flooding. The ISC annulus must fill with water before challenging the CCC seals. The CCC inside the ISC has one O-ring seal plus the inner metal seal that must fail. Water intrusion into the TRIGA casks was considered very unlikely. The TRIGA casks have a Helicoflex (Helicoflex is a trademark of the Helicoflex Corporation) metallic seal for long-term storage integrity. Also, the Rad-Vault (Rad-Vault is a trademark of Chem-Nuclear Systems, Inc) in which the TRIGA casks are stored has a lid designed to prevent rain intrusion. Water intrusion into the welded LWR canister was judged to be not credible. The leaktight container is analyzed not to fail during all DBAs. The NAC-1 casks (manufactured by Nuclear Assurance Corporation) are within a container that meets the standards of the International Standards Organization and provides weather protection. The LWR canister is welded closed.

[c] Cladding failure at a rate anticipated for commercial LWR fuel has been shown to have a negligible effect on the LWR fuel in SNF-4875, 1999, *Criticality Evaluation for Long Term Storage of Light Water Reactor Fuel in NAC-1 Casks*, Rev. 0. The zircaloy cladding will not suffer significant corrosion at the storage temperatures in the casks. A small percentage of pins might have cladding with breaches as a result of irradiation. These breaches could result in partial conversion of UO_2 to U_3O_8 in the affected pins, resulting in swelling and pin breakage. However, the number of pins affected is too small to significantly affect the k_{eff} of the assemblies. SNF-8924, 2001, *CSER 01-011: Criticality Safety Evaluation for Light Water Reactor Fuel in NAC-1 Casks*, and HNF-4832, 1999, *CSER 99-004: NFS-4/NAC-1 Spent Fuel Shipping Cask Criticality Safety Evaluation Report for Loose LWR Pins*, Rev. 0, also analyzed the LWR fuel assuming broken fuel pins and concluded that accidental criticality was incredible. TRIGA fuel is a metallic uranium hydroxide that is not susceptible to corrosion in water or air. The hot-cell rot phenomenon is unique to the FFTF fuel.

[d] The worst-case drop or crane collision has been analyzed for the ISCs in Chapter D3.0 and shown not to challenge the integrity of the CCC or ISA. However, the structural integrity of the fuel assemblies is not known and is assumed to fail in a drop. Also, hot-cell rot may occur in the FFTF fuel assemblies, so tipping the ISC could also result in fuel geometry changes. Crushing of the Rad-Vault and TRIGA casks has not been demonstrated incredible for the catastrophic impact of a crane failure. Effects of this event are analyzed in Section D6.3.3. The structural integrity of the LWR canisters and the LWR fuel rods will not be affected by the worst-case cask drop or natural phenomena hazard events, as discussed in Chapter D3.0.

[e] Misloading of Ident-69 containers into the center tube of a CCC was not considered credible, as the Ident-69 is too long to fit in the center tube without physical modification.

CCC	=	core component container.
DBA	=	design basis accident.
FFTF	=	Fast Flux Test Facility.
ISA	=	interim storage area.
ISC	=	interim storage cask.
LWR	=	light water reactor.

D6.3.3.1 Fast Flux Test Facility Fuel.

The k_{eff} of numerous normal and accident conditions for FFTF fuel stored in ISCs have been analyzed in the CSERs prepared for the storage of FFTF spent fuel: WHC-SD-FV792-DA-004, *Criticality Safety Evaluation of Ident 69s in Core Component Container*; WHC-SD-FF-CSER-002, *Criticality Safety Evaluation Report of Fuel Assemblies in Core Component Containers*; and WHC-SD-FF-CSER-004, *Criticality Safety Evaluation for Long Term Storage of FFTF Fuel in Interim Storage Casks*, Revisions 1, 1-A, and 1-B. The analyses in these documents are summarized in Sections D6.3.3.1.1 and D6.3.3.1.2.

D6.3.3.1.1 Analysis Models.

Criticality Safety Evaluation of Ident-69s in the Core Component Container

WHC-SD-FV792-DA-004 presents an analysis of k_{eff} in ISCs loaded with six Ident-69 pin containers, with the center tube of the CCC empty. The analysis was performed with the MCNP Monte Carlo program, version 3B, which is described in LA12625, *MCNP — A General Monte Carlo N-Particle Transport Code.* A parametric analysis was performed on the Ident-69 configuration for a flooded CCC. The following parameters were varied:

- Total number of pins
- Water density
- Square array vs. hexagonal array
- Number of CCC tubes loaded.

From the parametric analysis, the most reactive configuration for the Ident-69 was determined to be 97 pins in a square array. This analysis did not consider the effects of hot-cell rot. All cases relevant to ISA storage assume intact fuel pins. It is known that FFTF fuel lattices with a pin pitch less than 0.90 in. are under-moderated, thus the Ident-69 k_{eff} at the optimal pin pitch of ~0.5 in. corresponding to 97 pins was found to be most reactive when the CCC tubes were fully flooded with water. The optimal 97-pin configuration provides an upper bound on the k_{eff} of all flooded Ident-69 configurations. This configuration was used in the later CSERs (WHC-SD-FF-CSER-002; and WHC-SD-FF-CSER-004, Revs. 1, 1-A, and 1-B) to determine k_{eff} for flooded configurations reported in Table D6-4.

The assumptions used in the analysis are as follows:

- The fuel modeled is outer driver fuel, Type 4.1, with 29.28 wt% ^{239}Pu and ^{241}Pu and 11.63 wt% ^{240}Pu, which is the highest total plutonium enrichment for any of the FFTF fuel pins.
- Unirradiated fuel was modeled; no credit was taken for burnup.
- The radial dividers in the Ident-69 were neglected (resulting in a positive bias in k_{eff}).

The model was validated against experiments with fast test reactor fuel pins in water, as reported in "Critical Experiments with Fast Test Reactor Fuel Pins in Water" (Bierman et al. 1979). The validation showed a small positive bias in k_{eff} of ~ 0.006 for the MCNP results when compared to the benchmark experiments. In addition, omitting the radial dividers from the Ident-69 model introduced another positive bias of 0.013 in k_{eff}.

Table D6-4. Interim Storage Cask k_{eff} Analysis Summary Results. (4 sheets)

CCC loading[a]	Hazardous Condition(s) present				Upper bound on k_{eff}[c]	Reference
	Misloaded	Flooded[b]	Tipped	Hot-cell rot		
7 DFAs					0.4530	WHC-SD-FF-CSER-004, Rev. 0, pg. 12[d]
7 DFAs				X	0.9494	WHC-SD-FF-CSER-004, Rev. 0, pg. 12[d]
7 DFAs		X			0.7034	WHC-SD-FF-CSER-002[e]
7 DFAs		X		X	0.9905	WHC-SD-FF-CSER-004, Rev. 0, pg. 14[d] (duct failure assumed - allows fuel compaction to 13 in.)
7 DFAs		X	X	X	1.1553	WHC-SD-FF-CSER-004, Rev. 0, pg. 14[d] (duct failure assumed - fuel height 36 in.)
7 Ident-69s	X				0.4530	Bounded by 7 dry DFAs.
6 Ident-69s (center tube empty)					0.4530	Bounded by 7 dry DFAs.[f]
6 Ident-69s (center tube empty)		X			0.8859	WHC-SD-FF-CSER-004, Rev. 0, pg. 17[d]
6 Ident-69s (center tube empty)				X		Bounded by 7 DFAs with hot-cell rot.[f]
6 Ident-69s and center tube with DFA[g]	X	X			0.9693	WHC-SD-FF-CSER-004, Rev. 1, App. C, pg. 43[h]
6 Ident-69s and center tube with DFA[g]	X			X	0.9494	Bounded by 7 dry DFAs with hot-cell rot.[i]
5 Ident-69s mixed with DFAs filling remaining CCC tubes, with DFA in center tube		X		X	0.9498	WHC-SD-FF-CSER-004, Rev. 1, App. C, pg. 43[h]

Table D6-4. Interim Storage Cask k_{eff} Analysis Summary Results. (4 sheets)

CCC loading[a]	Hazardous Condition(s) present				Upper bound on k_{eff}[c]	Reference
	Misloaded	Flooded[b]	Tipped	Hot-cell rot		
5 Ident-69s mixed with DFAs filling remaining CCC tubes, with DFA in center tube		X			0.9394	WHC-SD-FF-CSER-004, Rev. 1, App. C, pg. 43[h]
5 Ident-69s mixed with DFAs filling remaining CCC tubes, with DFA in center tube				X	0.9494	Bounded by 7 DFAs with hot-cell rot.[i]
>1 Ident-69 mixed with DFAs filling remaining CCC tubes, with Ident-69 in center tube	X (no Ident-69s are allowed in center position)	X			1.0139	WHC-SD-FF-CSER-004, Rev. 1, App. C, pg. 42[h] (up to 3 Ident-69s were found to be acceptable, but were not allowed to simplify loading procedures).
6 metal fuel assemblies with center tube empty[j]					0.4136	WHC-SD-FF-CSER-004, Rev. 1-B[k]
6 metal fuel assemblies with center tube empty		X			0.8241	WHC-SD-FF-CSER-004, Rev. 1-B[k]
6 metal fuel assemblies with center tube empty				X	0.5064	Bounded by hot-cell rot case for 5 metal assemblies with 2 DFAs.
6 metal fuel assemblies with center tube empty		X	X	X	0.8964	WHC-SD-FF-CSER-004, Rev. 1-B[k] (36 in. fuel height caused by tipping, ducts intact).
>1 metal assembly mixed with DFAs filling remaining CCC tubes, with metal assembly in center tube	X (no metal assemblies allowed in center position)	X	X	X	0.9693	WHC-SD-FF-CSER-004, Rev. 1-B[k] (36 in. fuel height caused by tipping, ducts intact).

Table D6-4. Interim Storage Cask k_{eff} Analysis Summary Results. (4 sheets)

CCC loading[a]	Hazardous Condition(s) present				Upper bound on k_{eff}[c]	Reference
	Misloaded	**Flooded**[b]	**Tipped**	**Hot-cell rot**		
>1 metal assembly mixed with DFAs filling remaining CCC tubes, with metal assembly in center tube	X (no metal assemblies allowed in center position)	X			0.8999	WHC-SD-FF-CSER-004, Rev. 1-B[k]
>1 metal assembly mixed with DFAs filling remaining CCC tubes, with metal assembly in center tube	X (no metal assemblies allowed in center position)				0.4682	WHC-SD-FF-CSER-004, Rev. 1-B[k]
5 metal assemblies mixed with DFAs filling remaining CCC tubes, with DFA in center tube		X	X	X	0.9293	WHC-SD-FF-CSER-004, Rev. 1-B[k] (36-in. fuel height caused by tipping, ducts intact).
5 metal assemblies mixed with DFAs filling remaining CCC tubes, with DFA in center tube		X		X	0.7625	WHC-SD-FF-CSER-004, Rev. 1-B[k] (minimum fuel height, ducts intact).
5 metal assemblies mixed with DFAs filling remaining CCC tubes, with DFA in center tube				X	0.5064	WHC-SD-FF-CSER-004, Rev. 1-B[k] (minimum fuel height with ducts intact).
5 metal assemblies mixed with DFAs filling remaining CCC tubes, with DFA in center tube		X			0.8646	WHC-SD-FF-CSER-004, Rev. 1-B[k]

Table D6-4. Interim Storage Cask k_{eff} Analysis Summary Results. (4 sheets)

CCC loading[a]	Hazardous Condition(s) present				Upper bound on k_{eff}[c]	Reference
	Misloaded	Flooded[b]	Tipped	Hot-cell rot		
2 metal assemblies mixed with 5 Ident-69s	X (Ident-69s are not allowed with metal assemblies)	X	X	X	0.9961	WHC-SD-FF-CSER-004, Rev. 1-B[k] (36 in. fuel height caused by tipping, ducts intact).

[a] Table D6-4 addresses fully loaded CCCs. The k_{eff} for partially loaded CCCs are bounded by the fully loaded configurations.

[b] The flooded condition for storage of FFTF fuel in the ISC has been determined to be not credible.

[c] This number represents the k_{eff} calculated in the analysis, plus an additional margin to account for bias and uncertainty, as reported in the CSER.

[d] WHC-SD-FF-CSER-004, 1995, *Criticality Safety Evaluation for Long Term Storage of FFTF Fuel in Interim Storage Casks*, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

[e] WHC-SD-FF-CSER-002, 1993, *Criticality Safety Evaluation Report of Fuel Assemblies in Core Component Containers*, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

[f] Administrative controls on Ident-69 containers limit the number of pins to 217 (same as DFAs). However, Ident-69 containers could hold up to 233 pins. If all six Ident-69 containers are loaded with 233 pins, the total number of pins is still less than an ISC loaded with seven DFAs (1,398 pins vs. 1,519 pins). Also, the geometry of fuel in an ISC with one tube empty is less optimal for the hot-cell rot hazard than a fully loaded ISC. Therefore, the seven DFA case bounds the six Ident-69 case for hot-cell rot.

[g] Combinations of DFAs and Ident-69s were required to minimize the number of ISCs.

[h] WHC-SD-FF-CSER-004, 1996, *Criticality Safety Evaluation for Long Term Storage of FFTF Fuel in Interim Storage Casks*, Rev. 1, Westinghouse Hanford Company, Richland, Washington.

[i] This case was not analyzed in the CSERs and could contain slightly more fuel than an ISC loaded with seven DFAs, if the Ident-69 containers are overloaded with 233 pins. However, the Ident-69 pin containers are not expected to be impacted by hot-cell rot. An ISC loaded with DFAs and Ident-69 containers (with failed fuel but intact containers) is judged to be less reactive than an ISC loaded with seven DFAs subject to complete structural failure (fuel and duct failure) from hot-cell rot.

[j] Only six metal assemblies were made.

[k] WHC-SD-FF-CSER-004, 1997b, *Criticality Safety Evaluation for Long Term Storage of FFTF Fuel in Interim Storage Casks*, Rev. 1-B, Westinghouse Hanford Company, Richland, Washington.

CCC = core component container.
CSER = criticality safety evaluation report.
DFA = driver fuel assembly.
FFTF = Fast Flux Test Facility.
ISC = interim storage cask.

Criticality Evaluation for the Long-Term Storage of Fast Flux Test Facility Fuel in Interim Storage Containers (Rev. 0)

WHC-SD-FF-CSER-004 (Rev. 0), analyzed the k_{eff} of ISCs loaded with seven DFAs subject to hot-cell rot. The Monte Carlo criticality computer code, MONK6B, was used for the calculations of k_{eff}. Various geometries resulting from disintegration of the fuel due to hot-cell rot were studied, as follows:

- A set of parametric cases was analyzed that assumed complete fuel and cladding failure with intact assembly ducts. In this model, fuel and cladding were uniformly mixed within the intact duct. Heights from 21 in. to 36 in. for the uniform mixture were analyzed. Both dry and flooded configurations were analyzed.
- A case similar to the above cases but assuming failure of the assembly duct was also analyzed. The homogeneous mixture of fuel, cladding, and duct debris was found to have a minimum height of 12 in. Both dry and flooded configurations were analyzed. Fuel debris in pellet or powder form was also analyzed for the flooded configuration.
- A dry case was analyzed assuming that the cladding and duct debris were separated from the fuel debris, resulting in a minimum 7-in. fuel height.

The geometries analyzed provide an upper bound on k_{eff} for the possible configurations resulting from hot-cell rot in an ISC loaded with DFAs.

The assumptions used in the analysis are as follows:

- The fuel modeled is outer driver fuel, Type 4.1, with 29.28 wt% ^{239}Pu and ^{241}Pu and 11.63 wt% ^{240}Pu, which is the highest total plutonium enrichment for any of the FFTF fuel pins.
- Unirradiated fuel was modeled; no credit was taken for burnup.

The analysis also considered storage of multiple ISCs on a pad by analyzing an infinite planar array of adjacent, flooded ISCs. However, large scale flooding of the ISA is not considered credible in the hazard analysis discussed in Section D6.3.2.

In addition, the possibility of fuel heights larger than the nominal 36 in. was considered. These larger fuel heights were postulated as the result of fuel and cladding debris spreading out within the assembly duct in a horizontal ISC (e.g., due to tipping). Assembly ducts were assumed to be intact for the extended height cases.

Validation of the MONK6A computer code for WHC-SD-FF-CSER-003, *Criticality Evaluation for the Long Term Storage of FFTF Fuel in Interim Storage Casks*, can be applied to the MONK6B computer code. WHC-SD-SQA-CSWD-20019, *Monk Validation Study Comment Resolution*, demonstrated the equivalence of MONK6B to MONK6A, as the cross-section library of the two versions is identical. To validate the accuracy of the MONK6A model, WHC-SD-FF-CSER-003 made comparison calculations with the previous criticality analyses for intact DFAs in WHC-SD-FF-CSER-002. The only difference between the calculations was that the MCNP calculations in WHC-SD-FF-CSER-002 used explicitly modeled fuel pins, while the

MONK6A calculations used homogenized atom densities for the fuel pins and water. The MONK6A model was found to give a slightly larger k_{eff} that was within the statistical uncertainty of the calculation at the 95% confidence level. The validation document, WHC-SD-SQA-CSWD-20019, *MONK6A Pu Validation*, states that a calculated value of $k_{eff} < 0.935$ is required to assure $k_{eff} < 0.95$, including bias and uncertainty. Direct comparisons of MONK6A calculations against critical experiments show that calculated k_{eff} values less than 0.935 ensure that $k_{eff} < 0.95$, including uncertainty and bias at the 95% confidence level. For comparison to the safe subcritical limit of 0.95, a value of 0.015 was added to the calculated values of k_{eff} to obtain the values in Table D6-4.

Criticality Evaluation for the Long-Term Storage of Fast Flux Test Facility Fuel in Interim Storage Containers (Rev. 1)

WHC-SD-FF-CSER-004 (Rev. 1) adds Appendix C to Rev. 0. Appendix C analyzes combinations of DFAs and Ident-69s for flooded conditions. This analysis was performed to allow combinations of DFAs and Ident-69 containers to minimize the number of ISCs required in the storage area. The mixed configurations were analyzed using the same fuel assumptions discussed previously for the DFAs. All pin containers were assumed to be optimally loaded with 97 pins. The following cases were analyzed:

- One to seven Ident-69s with the remaining tubes occupied by DFAs. The center CCC tube is occupied by an Ident-69.
- Zero to five Ident-69s with one outer tube empty and the center tube with a DFA.
- Six Ident-69s in the outer tubes.
- Five Ident-69s in the outer tubes and two DFAs in the remaining tubes.

The analysis also considered storage of multiple ISCs on a pad, with combinations of Ident-69s and DFAs, by analyzing an infinite planar array of adjacent, flooded ISCs with five Ident-69s and two DFAs. However, large scale flooding of the ISA is not considered credible in the hazard analysis discussed in Section D6.3.2.

Criticality Evaluation for the Long-Term Storage of Fast Flux Test Facility Fuel in Interim Storage Containers (Rev. 1-A)

WHC-SD-FF-CSER-004 (Rev. 1-A) is written as Appendix D to WHC-SD-FF-CSER-004. This document contains analysis of the k_{eff} of modified Ident-69s containing returned FFTF experimental fuel and repackaged fuel in a spent fuel container, as described in Section D6.4.2. The experimental fuel is from the following FFTF test assemblies: ACO-3, ACO-1, FO-2, MFA-1, AAD-5, DE4-1, D9-1, DEA-2, D9-4 and C-1. The pins from each of these assemblies have previously been classified as equivalent to MOX fuel for the purposes of criticality safety. The MONK6B computer code was used to calculate k_{eff}. The code validation discussion for WHC-SD-FF-CSER-004 (Rev. 1) is applicable to Appendix D as well. The configurations of the flooded, water-reflected, modified Ident-69 analyzed are as follows:

- 48 stainless steel tubes inside a spent fuel canister with 4 kg of MOX fuel homogeneously distributed within each tube volume. Fuel within the tubes is dry, all other void spaces are flooded. Tube pitch within the spent fuel canister was varied from 0.50 in. (close packed) to 0.67 in. (spread out to the volume of Ident-69).

- 4 kg of MOX fuel in pellet form (0.494 cm diameter spheres) inside a flooded spent fuel canister. Pellet pitch was varied from 0.20 in. (close packed) to 0.45 in. (homogeneous distribution throughout spent fuel canister). The stainless steel of the fuel tubes was neglected.

- 4 kg of MOX fuel in pellet form (0.494 cm diameter spheres) inside a flooded, modified Ident-69. Pellet pitch was varied from 0.20 in. (close packed) to 0.88 in. (homogeneous distribution inside the modified Ident-69). The stainless steel of the spent fuel canister and the fuel tubes was neglected.

- 4 kg of MOX fuel in powder form inside a flooded, modified Ident-69. The stainless steel of the spent fuel canister and the fuel tubes was neglected.

The calculated k_{eff} values for these configurations were compared to a calculated k_{eff} value for a flooded, water-reflected, optimized Ident-69 loaded with 97 pins, as analyzed in WHC-SD-FF-CSER-004 (Rev. 1).

The following assumptions were made in the analysis:

- The modified Ident-69s contain less than 4 kg of experimental pins and repackaged fuel.

- Fuel pin cladding was ignored.

Criticality Evaluation for the Long-Term Storage of Fast Flux Test Facility Fuel in Interim Storage Containers (Rev. 1-B)

WHC-SD-FF-CSER-004 (Rev. 1-B) presents analysis of the k_{eff} of CCCs loaded with combinations of metal fuel assemblies and DFAs. The MONK6B computer code was used to determine k_{eff} values. The following cases were analyzed:

- One to seven metal fuel assemblies in a CCC with a metal assembly in the center tube and the remaining tubes occupied by DFAs. The ISC was assumed to be flooded. Fuel and cladding were assumed to be homogeneously distributed over a 36-in. height (nominal fuel height) within intact fuel assembly ducts.

- Four or five metal assemblies with a DFA in the center tube and the remaining tubes occupied by DFAs. Dry and flooded configurations were analyzed. Fuel and cladding were assumed to be homogeneously distributed over a 36-in. height (nominal fuel height) or over a 12-in. height (minimum height of collapsed fuel and cladding debris) within intact fuel assembly ducts. Dry and flooded configurations were analyzed. Several cases with dry and flooded intact fuel were also considered.

- Five metal assemblies with two DFAs (one in the center tube) within a flooded ISC. Fuel and cladding were assumed to be homogeneously distributed over 36-in. to 140-in. heights (nominal fuel height) within intact fuel assembly ducts. This case represents a horizontal ISC (e.g., due to tipping during movement or vehicle collision).

- Six metal assemblies with the center tube occupied or unoccupied. The ISC was assumed to be flooded. Fuel and cladding were assumed to be either intact or completely failed due to hot-cell rot. For the hot-cell rot cases, the fuel and cladding were homogeneously distributed over a 36-in. height (nominal fuel height) within intact fuel assembly ducts.
- Seven metal assemblies. The ISC was assumed to be flooded. Fuel and cladding were assumed to be homogeneously distributed over a 36-in. height (nominal fuel height) within intact fuel assembly ducts.

Several configurations having combinations of metal assemblies and Ident-69s were also analyzed, which assumed intact ducts, a flooded ISC, and the fuel and cladding homogeneously distributed over a 36-in. height due to hot-cell rot.

The analysis used the following assumptions:

- Metal fuel has 32.4 wt% 235U enrichment.
- Unirradiated fuel was used; no credit was taken for burnup.

Validation of the MONK6B computer code is discussed previously in the summary of WHC-SD-FF-CSER-004 (Rev. 1). For comparison to the safe subcritical limit of 0.95, a value of 0.015 for bias plus uncertainty was added to the calculated values of k_{eff} and reported in Table D6-4.

D6.3.3.1.2 Analysis Results.

Analysis results are summarized in Table D6-4. The table contains entries for all fully loaded CCC configurations and all credible hazardous conditions identified in the hazard evaluation. In addition, some CCC configurations that were found to be not credible are also presented, since they were analyzed in the CSERs. Some of these cases have k_{eff} values beyond the acceptable limit, but they do not require administrative or engineered controls since they result from hazardous conditions that were considered incredible in the hazard evaluation (Section D6.3.2).

Normal Conditions

Normal conditions consist of dry ISCs loaded with structurally intact DFAs, Ident-69s, metal assemblies, or modified Ident-69s subject to the restrictions described in Section D6.3.1.

From Table D6-4, the highest k_{eff} for normal configurations in the table (i.e., no hazardous conditions present) was calculated to be 0.4125, including bias and uncertainty, for an ISC loaded with six metal assemblies. Metal assemblies were found to be slightly more reactive than DFAs. This upper limit for normal configurations is well below the safe subcritical limit of 0.95.

For a dry ISC loaded with intact fuel, Ident-69s were found to be less reactive than DFAs. The k_{eff} for intact dry fuel increases with decreasing pin pitch. The pin pitch for the DFA is 0.284 in., while the pin pitch for 217 pins within an Ident-69 is greater than 0.30 in. (WHC-SD-FV792-DA-004). Thus, the most reactive dry configuration for Ident-69s with 217 intact pins is less reactive than a dry DFA with 217 pins.

Results of the analysis for the storage of multiple ISCs indicate that the CCCs are neutronically separated within the ISC. Compared to the k_{eff} of a single ISC, model results for an infinite planar array of adjacent ISCs showed an increase in k_{eff} on the order of the statistical uncertainty in k_{eff}. Thus, for arrays of dry ISCs with intact fuel, k_{eff} for the array will still fall well below the safe subcritical limit of 0.95.

Abnormal Conditions

As discussed in Section D6.3.2, individual hazardous events that were determined to be credible, but unlikely, are as follows:

- Misloading of a cask prior to transport to the ISA
- Tipping or dropping of a cask during handling (structural damage to the CCC due to a design basis drop was shown to be not credible in Chapter D3.0).

The results of the analysis in WHC-SD-FF-CSER-004 (Rev. 1-A) for experimental fuel in a modified Ident-69 container show that for all credible hazardous conditions, the k_{eff} for the modified Ident-69 will be bounded by k_{eff} for an optimized, unmodified Ident-69 container, as analyzed in WHC-SD-FF-CSER-004 (Rev. 1). Therefore, all hazardous ISC configurations with modified Ident-69 containers are bounded by the configuration obtained when the modified Ident-69 containers are replaced by optimized unmodified Ident-69 containers.

The hazard analysis also identified hot-cell rot as an anticipated hazard over the 40-year storage lifetime of the casks. The actual likelihood of hot-cell rot affecting the fuel pins is highly uncertain but has been observed at several other facilities. The thickness of the stainless steel in the fuel assembly ducts is several times greater than the fuel cladding itself and is less susceptible to corrosion than the fuel cladding. Ident-69 containers have not been exposed to thermal cycling and the corrosive environment of the sodium coolant during the cleaning process prior to loading in the ISCs and are not expected to suffer from the effects of hot-cell rot. However, in order to provide an upper bound on the likelihood of the hazard, the hazard analysis assumed that the event will occur and that fuel pins and/or assembly ducts may suffer complete structural failure. The CCC tubes were assumed to be unaffected by any corrosion mechanisms due to their robust design requirements. Hot-cell rot was modeled in the k_{eff} analyses by either (1) assuming that fuel and cladding was homogeneously mixed within the void space of the ducts, or (2) assuming that fuel, cladding, and assembly duct material were homogeneously mixed within the void space of the CCC tube. The height of the fuel and cladding debris, or fuel and cladding and duct debris, was varied parametrically in the k_{eff} calculations to determine the geometric configuration with the largest k_{eff}.

k_{eff} of Configurations Due to Hot-cell Rot

The largest value of k_{eff} (including bias and uncertainty at the 95% confidence level) for a dry, normally loaded ISC that has experienced hot-cell rot was found to be 0.9494. This extreme case of hot-cell rot was obtained for a CCC loaded with seven DFAs by assuming the entire assembly has disintegrated. In addition, the stainless steel of the cladding and ducts was assumed to have separated from the fuel debris by some unknown mechanism. The separated fuel was compacted to its minimum possible height of 7.2 in. within the bottom of the CCC tubes to form the most reactive configuration constrained only by the CCC tube geometry.

keff of Single Contingency Configurations

Single-contingency configurations are defined as those that occur due to a single unlikely hazardous event (with or without hot-cell rot), since some form of hot-cell rot was assumed to occur in the hazard analysis. Single-contingency events were assumed to include the possibility of hot-cell rot combined with one other unlikely hazardous event.

Misloaded cask – From Table D6-4, the most reactive misloaded cask configuration with intact fuel pins was found to be five metal fuel assemblies and two DFAs in the CCC, with a metal assembly in the center position (a configuration with six metal assemblies and one DFA was not analyzed, but should give a slightly higher k_{eff} – still far below 0.95). Metal assemblies are slightly more reactive than DFAs.

Note: Misloading of a cask with seven Ident-69 containers was considered to be not credible by the hazard evaluation and was not systematically analyzed in the CSERs. However, two cases with seven Ident-69 containers were analyzed and are reported in Table D6-4 for information only.

Tipped cask – Tipping or dropping a cask could result in damaged fuel that is spread out within the void space of the CCC tubes. The k_{eff} of a tipped cask with dry fuel is bounded by the analysis for hot-cell rot (discussed previously), since dry fuel has been shown to be more reactive when placed in the most compact geometry. Tipped or dropped casks with dry fuel were not analyzed in the CSERs and are not shown in Table D6-4. Tipping or dropping a flooded cask was analyzed in the CSERs and is discussed below under "Flooded Cask."

keff of Other Hazardous Conditions

The remaining hazardous conditions that have not been excluded as not credible by the hazard evaluation are as follows:

Misloaded and Tipped Cask – This configuration could result in damaged fuel spread out within the void space of the CCC tubes. The k_{eff} for the system is bounded by the misloaded configuration with hot-cell rot (discussed previously) since dry fuel was shown to be more reactive when compacted to its minimum possible height.

Incredible Hazardous Conditions

Hazardous conditions found to be not credible by the hazard evaluation, but were analyzed in the CSERs are as follows:

Flooded Cask – Cask flooding at the ISA was judged to be not credible by the hazard evaluation. However, flooded configurations analyzed in the CSERs are reported in Table D6-4 for completeness. The flooded configurations were analyzed in the CSERs to ensure that fuel remained safely subcritical during fuel handling operations, including washing activities.

From Table D6-4, the most reactive flooded, normally loaded cask configuration with intact fuel pins was found to be five metal fuel assemblies and two DFAs in the CCC, with a DFA in the center position. This configuration resulted in an upper bound on k_{eff} of 0.9380, including bias and uncertainty at the 95% confidence level. Including the possibility of hot-cell rot raises the k_{eff} slightly to 0.9453. The hot-cell rot configuration assumed that the fuel/cladding water debris extended over the normal height of the fueled region (36 in.) in order to achieve optimal moderation. In reality, hot-cell rot would tend to create debris that collects at the bottom of the CCC tube, which is a less optimally moderated geometry.

The infinite flooded array of ISCs containing five Ident-69s and two DFAs with intact fuel pins was calculated to have a k_{eff} near the safe subcritical limit in WHC-SD-FF-CSER-004 (Rev. 1). However, large scale flooding of the ISA is not considered credible in the hazard analysis discussed in Section D6.3.2.

Other flooded configurations were analyzed in the CSERs. Some of these configurations resulted in k_{eff} greater than the safe subcritical limit of 0.95. The highest k_{eff} found in the analysis occurred for a tipped, flooded ISC loaded with seven DFAs that have experienced complete failure due to hot-cell rot. The k_{eff} for this configuration was found to be 1.13. Additional criticality controls for flooded configurations for which $k_{eff} > 0.95$ are not necessary, as these conditions are not considered credible by the hazard evaluation.

D6.3.3.2 Neutron Radiography Facility TRIGA Fuel.

D6.3.3.2.1 Analysis Models.

The k_{eff} of normal and accident conditions for TRIGA fuel stored in the NRF TRIGA casks has been analyzed in WHC-SD-SQA-CSA-30006, *Criticality Safety Evaluation Report 95-012: Transfer of TRIGA Fuel from 308 Building Reactor Pool to Storage Casks*, WHC-SD-FF-CSER-006, *Criticality Safety Evaluation Report for TRIGA Fuel Storage at 400 Area Interim Storage Area*, and WHC-SD-TP-SARP-008. The models used in these documents are summarized below.

Safety Analysis Report for Packaging (Onsite) Neutron Radiography Facility TRIGA Packaging

WHC-SD-TP-SARP-008 contains the analysis of infinite planar arrays of TRIGA casks to show the casks meet the criticality requirements for transportation from the 300 Area to the 400 Area ISA at FFTF. As such, the explicit geometry and material of the Rad-Vault are not modeled. However, the infinite planar array of close-packed casks analyzed can be taken to represent an upper bound on casks in the Rad-Vault, which will be in a less reactive geometry.

The calculational model for MCNP included all components of a TRIGA cask loaded with 18 fuel elements. The cladding, fuel basket, inner container, and cask inner and outer shell with lead shielding were explicitly modeled. Normal fuel geometries inside dry cask arrays were analyzed with and without water occupying the interstitial space between the casks. The normal fuel geometry cases with no water occupying the space between casks represents an upper bound on the Rad-Vault configuration, since there will only be six fully fueled casks in the Rad-Vault, and the cask space will be larger than the arrays analyzed here.

In addition, abnormal fuel conditions were analyzed by removing the fuel cladding and stainless steel fuel baskets from the model. The fuel pitch inside the inner container was then varied to determine the worst-case geometry for the abnormal configuration (the geometry of the fuel elements themselves remained intact). The abnormal configurations were analyzed with or without water inside the casks or occupying the interstitial area between the infinite planar array of casks. A single case was analyzed with a homogeneous distribution of fuel and water within the region of the inner container.

The MCNP criticality computer code was used to perform the k_{eff} analysis for the infinite cask arrays. Uncertainty in the k_{eff} values for normal fuel geometry cases was estimated by combining the statistical uncertainty reported by MCNP with rough estimates for the error in cross-section, physical modeling, and dimensional tolerances. For normal configurations, this uncertainty in k_{eff} was determined to be 0.0036, which represents the statistical uncertainty in the Monte Carlo analysis (LA12625). For the abnormal cases, a configuration error was combined with the other error factors to account for the possibility that the limited parametric study performed on the fuel geometry and water density variations did not obtain the worst-case configuration. The overall uncertainty for abnormal cases was determined to be 0.0057. MCNP has been validated against experimental benchmarks by the code development group at Los Alamos National Laboratory (Whalen et al. 1991). The validation experiments included critical assemblies for low-enriched uranium systems, graphite and water-reflected systems, fast neutron systems (Godiva and Jezebel assemblies), and interactive (array) units. The code showed agreement to the experimental k_{eff} values within 1% for all benchmarks.

Criticality Safety Evaluation Report for TRIGA Fuel Storage at 400 Area Interim Storage Area

WHC-SD-FF-CSER-006 provides analysis of the abnormal fuel configuration resulting from the fall of a large crane onto the Rad-Vault.

The TRIGA cask safety analysis report for packaging (WHC-SD-TP-SARP-008) determined that the cask structural integrity would be maintained in the worst-credible drop or vehicle collision accident. However, it was postulated that a large crane falling onto the Rad-Vault could result in catastrophic structural failure of the vault and TRIGA casks, rearranging the fuel into a more reactive geometry than normal. With no geometry control, there is sufficient fuel in the Rad-Vault to go critical under idealized conditions.

The neutron transport code, WIMS-E,[5] which is documented in AEEW-R 1329, *A General Introduction to the Use of the WIMS-E Modular Program*, was used to parametrically vary the spacing of both the aluminum-clad and the stainless steel-clad TRIGA fuel elements in water. These heterogeneous lattice calculations were performed using the British Atomic Energy Authority 69-group nuclear cross-section library. Once the transport calculations were complete, WIMS-E used the 69-group radial flux solution to collapse cross-sections to a two-group set for the smeared lattice cell. The GOLF computer code, which has been validated and documented in WHC-SD-NR-SWD-024, *Software Certification Package for the GOLF Code*, then used these two-group cross-sections and performed neutron diffusion calculations to find the finite spherical and hemispherical radii required for $k_{eff} = 0.95$, given full water reflection.

The GOLF code has been validated for calculating critical dimensions for idealized geometries (such as spherical and hemispherical), when using nuclear cross-sections obtained from the WIMS-E lattice code. These validations are described in various sources, such as for

[5] WIMS is a trademark of Answers, the marketing organization of the United Kingdom Atomic Energy Authority.

uranium metal billets, N Reactor Mark IA critical experiments, lattices of uranium metal rods in water with various enrichments and rod sizes (including boron poisoned water lattices), and uranium and plutonium nitrate solutions. For the Mark IA critical experiments, WIMS-E and GOLF were used to predict the critical radius of the two-tier cylindrical arrangement that was measured to be critical. The critical prediction was slightly conservative (i.e., the predicted cylinder radius was less than what was measured as critical). The MCNP computer code was used to predict the k_{eff} of the WIMS-E/GOLF prediction ($k_{eff} = 0.99203 \pm 0.0017$). This establishes a WIMS-E/GOLF bias of 0.008, which was added to the calculated results for k_{eff} and then reported in Table D6-5. Statistical uncertainties in the WIMS-E/GOLF results were not reported in the analysis. Independent verification has been performed for WIMS-E/GOLF critical mass predictions using both the MCNP and MONK5 Monte Carlo computer codes and is documented in WHC-SD-NR-CSER-007, *Criticality Safety Evaluation Report, 105-KE Basin Fuel Encapsulation*, Appendices A and B; WHC-IP-0840, *Validation of WIMS-E for Prediction of Uranium, Plutonium Nitrate Solution Critical Masses*; and "Whole-Core Neutronics Modeling of a TRIGA Reactor Using Integral Transport Theory" (Schwinkendorf 1990).

The analysis assumed a worst-case random geometry of fuel occurred as a result of the accident. The most reactive geometry that could occur is a hemispherical pile of homogeneously distributed fuel and cladding. This event is judged to be incredible.

Criticality Safety Evaluation Report 95-012: Transfer of TRIGA Fuel from 308 Building Reactor Pool to Storage Casks

WHC-SD-SQA-CSA-30006 contains analysis of the k_{eff} of normal and hazardous conditions that may be encountered during transfer operations from the TRIGA reactor pool into TRIGA casks. The analysis is directed at the transfer operations and does not explicitly consider hazards associated with long-term storage of the fuel. Geometric configurations with the Rad-Vault are not directly analyzed. However, the case analyzed for a close-packed array of seven normally loaded casks can be taken as an upper bound on the k_{eff} of the actual Rad-Vault configuration. The cask array was modeled as a normally configured, dry cask with 18 aluminum clad or stainless steel clad fuel elements inserted in the aluminum fuel basket.

The MONK6B computer code was used to perform the k_{eff} analysis for the cask array. The documentation for the code includes results of validation calculations for several critical uranium-enriched systems. The benchmark experiments were performed on highly enriched systems or low-enriched systems, as opposed to the 20% enriched TRIGA fuel. WHC-SD-SQA-CSA-30006 judged the validity of the code for TRIGA fuel based on the benchmark results for highly and low-enriched critical experiments. The results of the validation calculations show a consistent positive bias in k_{eff} for the MONK6B calculations. This positive bias was not included in the reported values for k_{eff} in Table D6-5. Statistical uncertainties at the 2σ level were added to the calculated values of k_{eff}, and the result was reported in Table D6-5.

Table D6-5. Neutron Radiography Facility TRIGA Cask k_{eff} Analysis Summary Results.

TRIGA cask loading	Hazardous Condition(s) Present			Upper Bound on k_{eff}	Reference
	Flooded	Tipped	Crushed		
18 SS fuel pins		X		0.799	WHC-SD-TP-SARP-008[a] (optimal fuel geometry without clad)
18 AL fuel pins				0.4656	WHC-SD-SQA-CSA-30006[b] (7 cask array)
18 SS fuel pins				0.5818	WHC-SD-SQA-CSA-30006[b] (7 cask array)
18 SS fuel pins				0.7	WHC-SD-TP-SARP-008[a] (infinite planar array)
18 SS fuel pins	X	X		0.939	WHC-SD-TP-SARP-008[a] (optimal fuel geometry without clad)
18 AL fuel pins	X		X	0.858	WHC-SD-FF-CSER-006[c] (cask and vault crushed, fuel optimally moderated, inner cask steel co-mingled)
18 SS fuel pins	X			0.7564	WHC-SD-SQA-CSA-30006[b] (7 cask array, outer casks dry, central cask flooded)

[a] WHC-SD-TP-SARP-008, 2002, *Safety Analysis Report for Packaging (Onsite) NRF TRIGA Packaging,* Rev. 0-C, Fluor Hanford, Incorporated, Richland, Washington.

[b] WHC-SD-SQA-CSA-30006, 1995, *Criticality Safety Evaluation Report 95-012: Transfer of TRIGA Fuel from 308 Building Reactor Pool to Storage Casks,* Rev. 0, Westinghouse Hanford Company, Richland, Washington.

[c] WHC-SD-FF-CSER-006, 1995, *Criticality Safety Evaluation Report for TRIGA Fuel Storage at 400 Area Interim Storage Area,* Rev. 0, Westinghouse Hanford Company, Richland, Washington.

AL = aluminum.
SS = stainless steel.

D6.3.3.2.2 Analysis Results.

Analysis results for k_{eff} of the TRIGA casks are summarized in Table D6-5. The table contains entries for all cask configurations and all combinations of hazardous conditions identified as credible in the hazard evaluation. In addition, some cask configurations that were considered not credible by the hazard evaluation are also presented for information, since they were analyzed in the CSERs.

Normal Conditions

Normal conditions for the TRIGA casks consist of six TRIGA casks loaded with up to 18 intact dry fuel assemblies. Normal configuration analyses were performed for arrays of close-packed casks or an infinite lattice of close-packed casks. This geometry provides an upper bound on k_{eff}, since the actual geometry in the Rad-Vault includes extra spacing and absorber material due to the DOT-6M casks and the empty 55-gal. drum inside the vault. Inclusion of the DOT-6M casks would lower the k_{eff} compared to the analyzed geometries.

Analysis in WHC-SD-SQA-CSA-30006 found that the upper bound on k_{eff} for an array of seven normally loaded casks was 0.5818. The analysis shows that all normal conditions for casks loaded in the Rad-Vault have an upper bound on k_{eff} (including bias and uncertainty at the 95% confidence level), which is well below the safe subcritical limit of 0.95.

Abnormal Conditions

Hazardous conditions resulting from a single credible but unlikely hazardous event for the TRIGA fuel are represented by the following:

- Flooded configurations
- Tipped/dropped configurations with fuel rubble in abnormal geometries due to fuel pin failure
- Crushed configurations with fuel and/or cask rubble in abnormal geometries due to catastrophic failure of the Rad-Vault and casks.

Various combinations of these hazardous events were analyzed.

Note: Misloaded casks are not possible as there are no loading restrictions on the type of fuel elements loaded into the casks, and the fuel basket is to be fully loaded with the maximum number of elements.

Single-Contingency Hazardous Conditions

Single-contingency configurations are defined as those due to a single unlikely hazardous event.

Flooded Casks – The worst-case flooded cask condition analyzed in the CSERs was found to have an upper bound on k_{eff} of 0.7564, including bias and uncertainty at the 95% confidence level. The configuration analyzed in WHC-SD-SQA-CSA-30006 was a seven-cask array of fully loaded casks surrounded by water. Only the central cask in the array was flooded, with the outer six casks dry inside, which does not represent the worst-case flooding configuration. However, k_{eff} for a configuration with all seven casks flooded is bounded by the flooded and tipped configuration discussed below, which assumed an infinite planar array of casks and neglected fuel cladding and fuel basket stainless steel. The k_{eff} for the worst-case tipped and flooded hazardous configuration was found to fall below the safely subcritical limit, and, therefore, all configurations with flooding alone will be safely subcritical.

Tipped/Dropped Casks – The worst-case cask configuration resulting from fuel damage due to cask tipping or dropping was analyzed in WHC-SD-TP-SARP-008. The configuration analyzed was an infinite planar array of dry casks fully loaded with fuel pins. Cladding and fuel basket materials were neglected to represent the effects of fuel damage. The upper bound on k_{eff} for this configuration was found to be 0.799, which falls below the safe subcritical limit of 0.95.

Crushed Casks – The worst-case crushed cask configuration found in Table D6-5 gave a k_{eff} of 0.858. The configuration analyzed in WHC-SD-FF-CSER-006 was an infinite planar array of optimally moderated, close-packed, intact fuel rods. The stainless steel of the inner canister was co-mingled with the fuel. This infinite planar array configuration provides an upper bound on the k_{eff} of actual configurations that might result from the crushing of the Rad-Vault and dispersal of its contents. Separation of the inner canister stainless steel was not considered credible, but was analyzed in the CSER and is discussed below. This idealized debris configuration indicates that actual configurations resulting from crushing of the Rad-Vault will have a k_{eff} below the safe subcritical limit of 0.95.

Other Credible Hazardous Conditions

Tipped/Dropped and Flooded Casks – The worst-case cask configuration resulting from a flooded Rad-Vault and casks with fuel damage due to tipping or dropping was analyzed in WHC-SD-TP-SARP-008. The configuration analyzed was a flooded infinite planar array of internally flooded casks fully loaded with fuel pins. Cladding and fuel basket materials were neglected to represent the effects of fuel damage. The upper bound on k_{eff} for this configuration was found to be 0.939, which falls below the safe subcritical limit of 0.95.

Crushed and Flooded Casks – The hazardous configuration resulting from a combination of crushed and flooded casks is bounded by the analysis for crushed casks (discussed previously) since that analysis already assumed optimal moderation in the debris pile.

Incredible Hazardous Conditions

The most reactive abnormal condition analyzed was a hemispherical pile of fuel rubble resulting from a crane falling onto the Rad-Vault. The results of the analysis in WHC-SD-FF-CSER-006 showed that it is theoretically possible to exceed $k_{eff} = 0.95$ under optimal conditions of geometry and moderation within the rubble pile. This assumes that all absorbing material in the cask and Rad-Vault debris is somehow preferentially separated from the fuel—an event that is considered extremely unlikely. WHC-SD-FF-CSER-006 also showed that if the steel inner-liner in the cask is included in the rubble pile, the k_{eff} for an infinite planar array of TRIGA fuel is < 0.85. The worst-case scenario consisting of a crane failure, an optimal geometry of the fuel rubble, and separation of absorbing material from the cask is not considered credible. Further engineered or administrative controls to reduce the likelihood of this criticality event are not required.

D6.3.3.3 Commercial Light Water Reactor Fuel.

D6.3.3.3.1 Analysis Models.

The k_{eff} of normal and accident conditions for storage of commercial LWR fuel in NAC-1 casks has been analyzed in SNF-4875, *Criticality Evaluation for Long Term Storage of Light Water Reactor Fuel in NAC-1 Casks*, SNF-8924, *CSER 01-011: Criticality Safety Evaluation for Light Water Reactor Fuel in NAC-1 Casks*, and HNF-4832, *CSER 99-004: NFS-4/NAC-1 Spent*

Fuel Shipping Cask Criticality Safety Evaluation Report for Loose LWR Pins. The models used in these documents are summarized in the subsections that follow.

Criticality Evaluation for Long-Term Storage of Light Water Reactor Fuel in NAC-1 Casks

SNF-4875 contains analysis of the k_{eff} of NAC-1 casks loaded with PWR assemblies. Most of the materials and geometry of the NAC-1 casks loaded with PWR fuel assemblies from Calvert Cliffs and Point Beach were explicitly modeled in the analysis. The model included the cask inner liner, lead shielding, outer liner, steel lid and cask bottom, and neutron shield tanks. The cask lid and bottom thickness differed slightly from the as-built configuration. This dimension will not affect the k_{eff} calculations, since the cask is nearly an infinite system in the axial direction, and end effects are negligible. The inner canister was modeled with the design thickness, but the steel and aluminum in the inner canister ribs and the structural supports for the fuel assembly were neglected. The model did not take credit for the extra steel and spacing between casks provided by the International Standards Organization (ISO) containers, in which each cask will be stored. The fuel was modeled with the following input assumptions:

- Unirradiated fuel was modeled; no credit was taken for burnup.
- The zircaloy cladding was neglected.
- Enrichment for all pins in the model was taken as the highest actual value for any single pin in the assembly.
- The casks containing the Point Beach assemblies have a maximum of 179 pins, with 3.2 wt% enrichment.
- The cask containing the Calvert Cliffs full assembly has a maximum of 176 pins, with 2.72 wt% enrichment.
- The cask containing the Calvert Cliffs partial assembly has a maximum of 144 pins, with 2.73 wt% enrichment.

To determine an upper bound on six normal cask configurations, an infinite planar array of close-packed casks with dry inner containers and undamaged fuel was calculated. Moderation in the neutron shield tanks and in the interstitial region between the casks was varied to determine the most reactive configuration, and this configuration was analyzed to determine the upper bound on k_{eff} for the planar array.

The accident condition analyzed for the casks was a flooding scenario. As discussed in Section D6.3.2, hot-cell rot and fuel rearrangement due to tipping or collisions was not considered credible for the NAC-1 casks. Also, vertically stacked arrays were not considered credible. To determine the most reactive flooded configuration, the analysis focused on a single cask flooded internally and surrounded by water reflection. The number of pins in the cask was varied from the maximum of 196 to determine the optimal number. For a given number of pins, the pin pitch was chosen such that the square array filled the entire assembly cross-section. Once the optimal pin number was determined, the upper bound on k_{eff} (including bias and uncertainty) of the most reactive configuration was calculated. The optimal configuration for each fuel type was then used to analyze an infinite 3-dimensional array of close-packed casks, as for the dry case. Water density between the casks was varied to determine the most reactive configuration. The most reactive configuration was then analyzed to determine the upper bound on k_{eff}.

Several computer codes were used to model the fuel assemblies in the casks, to generate a cell-weighted cross-section, and to model the cask geometry, predict k_{eff}, and to perform scoping calculations to determine the optimal geometric configuration of the fuel in the flooded cask arrays to determine the optimal moderation between casks in the dry array calculations.

Uncertainty due to dimensional tolerances and statistical uncertainty were included in the final k_{eff} values reported in Table D6-6. Dimensional uncertainties were included using a one-sided tolerance limit factor, as outlined in "A Systematic Approach to Establishing Criticality Biases" (Larson 1995). For example, the uncertainty in the fuel pellet enrichment, density, and diameter was added to the values reported in Chapter D2.0, which represents an increase in the amount of fissile material to account for uncertainty in input parameters for the fuel. In the single cask calculations, the neutron reflection from the cask structure was increased by adding the dimensional tolerances to the thickness of the lead shielding. On the other hand, the infinite planar array calculations subtracted the dimensional tolerances from the cask dimensions to maximize interaction between the casks.

Code bias in calculations also was included in the k_{eff} values reported in Table D6-6 for the flooded arrays. Bias was not included in the dry array calculations. Since the k_{eff} values are so far below critical for the dry arrays, extrapolation of code bias from critical benchmark experiments would not be justified, and a criticality is incredible since the enrichment of the fuel is less than 5 wt% U^{235} (LA12808).

To validate the calculations, critical benchmark experiments were chosen that used UO_2 fuel pins with enrichments bounding those for the LWR fuel pins loaded in the NAC-1 casks. From the comparison results to the benchmark experiments, the code bias in k_{eff} was determined to be 0.004, with a standard deviation of 0.007. This bias was included in the upper bound k_{eff} values in Table D6-6. The computer code used to determine the most reactive configurations was validated by direct comparison with another computer code for optimization of the Point Beach fuel, single cask analysis. Both codes indicated that the 179-pin configuration was the most reactive.

Criticality Safety Evaluation for Light Water Reactor Fuel in NAC-1 Casks

SNF-8924 (CSER 01-011) was issued as an addendum to the CSER issued as SNF-4875 and extends the criticality safety evaluation to include broken fuel pins for UO_2 PWR fuel assemblies. This analysis credits an internal basket within the LWR canister. The design has an 8.25-in. x 8.25-in. cavity of a weldment created by 0.120-in thick stainless steel, with aluminum centering plates that extend out to a 11.875-in O.D. This 11.875-in. O.D. weldment fits inside the LWR canister, which is a 12-in. schedule 40 pipe (nominal 11.938-in I.D. and 12.750-in O.D.). Even with water moderation, accidental criticality is incredible when the basket and canister configuration is credited for geometry control. In addition, the conclusion also states, "If water moderation (flooding) is not a credible event, then a criticality is not credible even if the fuel were not constrained by the inner basket weldment." Water intrusion into the double welded LWR canister is considered incredible during storage at the ISA since the canister is leaktight. In addition, the ISA is above credible river flood levels and rain scenarios will not result in flash floods or freestanding water of depths required to flood the NAC-1 cavity.

Table D6-6. NAC-1 Cask k_{eff} Analysis Summary Results.

NAC-1[a] cask loading	Hazardous condition(s) present: Flooded	Upper bound on k_{eff}	Reference
Point Beach assembly		0.258	SNF-4875[b]
Calvert Cliffs full PWR assembly		0.244	SNF-4875[b]
Calvert Cliffs partial PWR assembly		0.260	SNF-4875[b]
Point Beach assembly	X	0.947	SNF-4875[b]
Calvert Cliffs full PWR assembly	X	0.937	SNF-4875[b]
Calvert Cliffs partial PWR assembly	X	0.948	SNF-4875[b]
Loose fuel pins		0.161	HNF-4832[c]
Loose fuel pins	X	0.630	HNF-4832[c]
Loose fuel pins	X (plus no inner pipe)	0.976	HNF-4832[c] (double contingent case)

[a] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

[b] SNF-4875, 1999, *Criticality Evaluation for Long Term Storage of Light Water Reactor Fuel in NAC-1 Casks*, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

[c] HNF-4832, 1999, *CSER 99-004: NFS-4/NAC-1 Spent Fuel Shipping Cask Criticality Safety Evaluation Report for Loose LWR Pins*, Rev. 0, Fluor Daniel Northwest, Incorporated, Richland, Washington.

PWR = pressurized water reactor.

NAC-1 Spent Fuel Shipping Cask Criticality Safety Evaluation Report

HNF-4832 analyzes the k_{eff} of a NAC-1 cask loaded with loose fuel pins from the 324 Building B-Cell Safety Cleanout Project. The original analysis evaluated an alternate configuration of LWR canister internals and a loose pin container that was fabricated from stainless steel pipe. The alternate configuration will not be used. HNF-4832, Rev. 0A, included an addendum that evaluated the loose pins loaded within a square inner container without limits on the condition, distribution, or spacing of the pins. One hundred eighteen loose fuel pins are to be loaded into a stainless steel rod consolidation container intended to restrict the pin spacing for the purpose of criticality safety. The consolidation container is an 8-in. x 8-in. square box with an overall length of 168 in. The original analysis of the k_{eff} assumed that the inner pipe of the old design containing the fuel pins had a circular cross-section and the steel plate welded to two half cylinders was neglected giving a more reactive configuration in the model than in the actual design. The rod consolidation box geometry is bounded by the original analysis assumption of a 10.9-in. cylindrical geometry.

The materials of the NAC-1 cask were explicitly modeled, including the rod consolidation container of stainless steel, the lead shielding, the outer stainless steel of the neutron shield tanks, and the top and bottom of the cask. The fuel was modeled with the following assumptions:

- Unirradiated fuel was modeled; no credit was taken for burnup
- Loading of the cask is limited to 17 PWR pins with enrichment < 3.04%
- Loading of the cask is limited to 101 BWR pins with enrichment < 2.93%
- The zircaloy cladding was included in the model.

The k_{eff} analysis (Rev. 0) considered a normal dry cask configuration with the inner pipe containing close-packed, dry fuel pins. Water reflection around the cask was assumed. The analysis also considered abnormal flooded conditions with water inside the cask and inner pipe. In addition, abnormal configurations were analyzed for a flooded cask without the inner pipe constraining the fuel. In these cases, fuel pitch within the inner cask region was varied to determine the optimal k_{eff}. The abnormal cases with no inner pipe were included to examine whether or not the inner pipe constraining the fuel geometry was required to fulfill double contingency for a flooded cask (flooding being the first contingency).

The MCNP computer code was used to determine k_{eff}. A series of benchmark experiments for low-enriched uranium systems was used to determine that the bias in k_{eff} for this analysis was 0.004. A standard value of 0.005 was used for the standard deviation in k_{eff} due to input uncertainties (e.g., dimensional tolerances, enrichment error, etc.). These uncertainties were combined with the statistical error in the Monte Carlo population to determine an error in the calculated k_{eff} of 0.011, including bias and uncertainty at the 95% confidence level. This value was added to all calculated values of k_{eff} and the total was reported in Table D6-6.

D6.3.3.3.2 Analysis Results.

Original analysis (Rev. 0) results for k_{eff} of the NAC-1 casks are summarized in Table D6-6. The table contains entries for all cask configurations and all credible hazardous conditions. Addendum 1 (Rev. 0A) extends the original analysis and concludes that within the geometry established by the rod consolidation container, the system reactivity will not exceed a k_{eff} of 0.95 and shows qualitatively that due to the form or distribution of fissile material, an accidental criticality is incredible. The distribution of fissile material contained by the square rod consolidation container of maximum 8.0-in. by 8.0-in. dimension supports the incredibility arguments.

Normal Conditions

Normal conditions for the NAC-1 casks consist of five casks with dry intact fuel in ISO containers stored side-by-side on the ISA pad. The most reactive configuration for dry cask arrays was found to have $k_{eff} = 0.25966$ for the Calvert Cliffs partial assembly in an infinite planar array. From the parametric study of optimal moderation, a completely dry array was found to be most reactive. This result represents an upper bound on k_{eff} for the actual NAC-1 cask array with three Calvert Cliffs assemblies, two Point Beach assemblies, and one cask loaded with odd pins. The normal configuration for the NAC-1 casks has a k_{eff} well below the safe subcritical limit of 0.95.

Abnormal Conditions

The only unlikely but credible hazardous event for the NAC-1 casks is flooding, which is presented below as a single unlikely contingency event.

Single-Contingency Hazardous Conditions

Flooding – From Table D6-6, the most reactive flooded configuration of NAC-1 casks was found to be an infinite planar array of Calvert Cliffs partial assemblies that resulted in $k_{eff} = 0.948$, including bias and uncertainty at the 95% confidence level. This array represents an upper bound on the k_{eff} of the actual NAC-1 cask configuration. Thus, the k_{eff} of all credible single-contingency hazardous configurations for the NAC-1 casks falls below the safe subcritical limit of 0.95. It is noted that flooding of the welded LWR containers is now considered incredible, as documented by the design basis accident (DBA) analyses that demonstrate that failure of the LWR canister will not occur, protecting its leaktight configuration.

Other Credible Hazardous Conditions

For the loose fuel pins, a flooded cask with no rod consolidation container constraining the geometry of the fuel resulted in an upper bound on k_{eff} of 0.976. This credible hazardous condition exceeds the safe subcritical limit of 0.95. To control this hazard, all fuel was analyzed within a cylindrical geometry with a diameter no greater than 10.9 in. within the cask, and the component(s) or structure providing this constraint must be qualified to maintain its structural integrity to the same degree as the casks. With fuel in the square rod consolidation container with a diagonal less than 10.9 in., the maximum k_{eff} for the system is well below the safe subcritical limit of 0.95. This control is discussed further in Section D6.4 and Chapter D4.0.

D6.4 CRITICALITY CONTROLS

D6.4.1 Engineered Controls

D6.4.1.1 Fast Flux Test Facility Fuel.

The geometry of the CCC was assumed to be intact in the criticality hazard analyses. Thus, no k_{eff} analyses were performed for double batching (the CCC geometry is assumed to prevent this hazard) or rearrangements of the CCC tubes due to structural failure. To protect the assumptions in Section D6.3.3 for the k_{eff} analysis of hazardous conditions, the ISC and the CCC are designated as system geometry controls that must be designed to maintain structural integrity for all design basis accidents (e.g., dropping, collision, etc.), including the seismic event. The design classification of the CCC and ISC structures and their detailed design requirements are discussed in Chapter D4.0.

D6.4.1.2 Commercial Light Water Reactor Fuel.

From the results of the analysis in HNF-4832, the geometry constraint of the rod consolidation container containing the loose fuel pins is required in order to provide double-contingency protection against criticality due to flooding of the cask. Therefore, the rod consolidation container is a safety-class component that must be designed and structurally qualified to the same quality level as the cask. The design classification of the rod consolidation container containing the fuel and its detailed design requirements are discussed in Chapter D4.0.

The square lattice geometry of the fuel assemblies was assumed to be intact by the original analysis in SNF-4875. Thus, no k_{eff} analyses were originally performed for fuel rubble or rearrangements of fuel within the inner canister due to structural failure. To protect the assumptions in Section D6.3.3 for the k_{eff} analysis of hazardous conditions, the NAC-1 cask inner container and fuel assemblies were designated as system geometry controls that must be designed and maintained to ensure structural integrity for all design basis accidents (e.g., dropping, collision, etc.), including the seismic event. Supplemental analysis in SNF-8924 evaluated the LWR inner container weldment (basket) for geometry control and determined that damaged fuel could be safely stored when the basket was credited. The fuel assembly basket is therefore designated as safety class. Another conclusion of SNF-8924 was that geometry control was not required even with damaged fuel, if water moderation (flooding) was not a credible event. The accident analysis in Chapter D3.0 demonstrates that the leaktight integrity of the LWR canister is maintained for all DBAs, and flooding is therefore considered incredible. The leaktight LWR canister is therefore designated as safety class. The design classification of the cask and fuel assembly structures and their detailed design requirements are discussed in Chapter D4.0.

D6.4.2 Administrative Controls

D6.4.2.1 Fast Flux Test Facility Fuel.

The Ident-69 analysis presented in WHC-SD-FV792-DA-004 makes assumptions regarding the fuel form and enrichment that are protected by administrative controls to ensure that all ISC configurations remain within the analyzed envelope. Specifically, ISCs containing fuel pins having greater than 29.28 wt% plutonium or any other experimental pins not analyzed in Section D6.3.3 cannot be accepted at the ISA without further analysis.

The criticality analysis for metal fuel assemblies in WHC-SD-FF-CSER-004 (Rev. 1-B) did not systematically analyze combinations of Ident-69 containers with metal assemblies in a CCC. Also, the CSER did not analyze certain misloaded ISCs containing metal assemblies structurally compromised by hot-cell rot. To ensure that configurations not analyzed by the CSER cannot exist at the ISA, acceptance of ISCs containing metal assemblies is prohibited by an administrative control until further analysis has been performed on metal assemblies with hot-cell rot and with combinations of Ident-69 containers and metal assemblies.

The criticality analysis in WHC-SD-FF-CSER-004 (Rev. 1-A) for repackaged fuel material and experimental pins assumed that not more than 4 kg of repackaged fuel and pins were contained in the spent fuel container loaded in a modified Ident-69 container. Also, as discussed in Section D6.3.3, the analysis considered only experimental pins that have been classified as MOX fuel, with not more than 31 wt% plutonium enrichment. To ensure that the existing analyses envelope all ISC configurations, these assumptions are protected by administrative controls. Specifically, the following ISC configurations cannot be accepted at the ISA without further analysis:

- ISCs containing a modified Ident-69 with more than 4 kg of repackaged fuel material and experimental pins.

- ISCs containing repackaged fuel material and experimental pins that are not packaged in stainless steel tubes packed within a spent fuel container inside a modified Ident-69, as described in Section D6.3.2.
- ISCs with experimental fuel not classified as MOX or having greater than 31 wt% plutonium.
- ISCs containing fuel pieces that may be mounted in moderating material (e.g., plastic).

The criticality analysis results in Section D6.3.3 indicate that a failure of administrative controls at FFTF, resulting in any other type (i.e., other than misloading a modified Ident-69 with experimental fuel and repackaged fuel, or misloading of an ISC with metal assemblies as described previously) of misloaded ISC that is transferred to the ISA, will not create a criticality concern at the ISA. All other credible accident scenarios associated with misloading at FFTF were shown to be safely subcritical and satisfy the double-contingency principle, without the requirement for additional administrative controls on cask receipt at the ISA.

D6.4.2.2 Commercial Light Water Reactor Fuel.

The criticality analysis for loose fuel pins showed that it is necessary to load the fuel into the consolidation container geometry for double-contingency protection against flooding. Therefore, a NAC-1 cask with loose fuel pins cannot be accepted at the 200 Area ISA unless the pins are loaded within a rod consolidation container, as described in Chapter D2.0.

D6.4.3 Application of Double Contingency

A general discussion regarding how the double-contingency principle is applied is provided in Section 6.4.3 of HNF-3553, Volume 1.

D6.4.3.1 Fast Flux Test Facility Fuel.

Table D6-4 shows the largest k_{eff} found in the analysis for normal cask configurations was 0.45 for five metal fuel assemblies and two DFAs in the CCC. Thus, all normal cask configurations have an upper bound k_{eff} well below the safe subcritical limit of 0.95.

For those hazardous configurations that result from, at most, a single unlikely contingency event (i.e., abnormal conditions that do not satisfy double contingency, such as misloading of a dry cask, etc.), the maximum k_{eff} is found to be 0.9494 for a CCC loaded with seven DFAs that have experienced complete failure due to hot-cell rot. Thus, all abnormal cask configurations that do not result from two or more unlikely, independent events have an upper bound k_{eff} below the safe subcritical limit of 0.95.

Some of the hazardous configurations in Table D6-4 that result from two or more unlikely contingency events have an upper bound on k_{eff} that exceeds the safe subcritical limit of 0.95. Notably, the hazardous configuration with a flooded and tipped CCC loaded with seven DFAs that have failed due to hot-cell rot was found to have a k_{eff} upper bound of 1.155. Not all hazardous configurations resulting from two or more unlikely events were analyzed in the CSERs, so there may be other conceivable hazardous configurations that have an even higher upper bound on k_{eff}, as calculated by the analysis scheme in the CSERs. Since these cases already satisfy the double-contingency principle, further controls to prevent criticality are not required in order to satisfy the double-contingency principle.

D6.4.3.2 Neutron Radiography Facility TRIGA Fuel.

Table D6-5 shows the largest k_{eff} found in the analysis for normal cask configurations was 0.5818 for stainless steel fuel assemblies in the Rad-Vault. Thus, all normal cask configurations have an upper bound k_{eff} well below the safe subcritical limit of 0.95.

The maximum k_{eff} for single-contingency hazardous events affecting the TRIGA casks was found to be 0.858 for the crushed Rad-Vault configuration represented by the infinite fuel assembly array analyzed in WHC-SD-FF-CSER-006. Thus, all abnormal Rad-Vault configurations that do not result from two or more unlikely, independent events have an upper bound k_{eff} that is below the safe subcritical limit of 0.95.

D6.4.3.3 Commercial Light Water Reactor Fuel.

Table D6-6 shows that the largest k_{eff} found for normal cask configurations was 0.260 for the Calvert Cliffs partial assembly. Thus, all normal NAC-1 cask configurations have an upper bound k_{eff} well below the safe subcritical limit of 0.95.

The maximum k_{eff} for single-contingency hazardous events affecting the NAC-1 casks was found to be 0.948 for the flooded Calvert Cliffs partial assembly. Thus, all abnormal NAC-1 cask configurations that do not result from two or more unlikely, independent events have an upper bound k_{eff} that is below the safe subcritical limit of 0.95.

One double-contingent event analyzed in HNF-4832 (i.e., flooded cask with loose fuel pins not constrained by a consolidation container) was found to have an upper bound on k_{eff} exceeding the safe subcritical limit of 0.95. The consolidation container containing the fuel within the inner canister was designated as a geometry control to ensure that double contingency is satisfied for NAC-1 cask flooding events. With the consolidation container intact, k_{eff} was analyzed to be below the safe subcritical limit of 0.95 and flooding is incredible.

D6.5 CRITICALITY PROTECTION PROGRAM

Section 6.5 of HNF-3553, Volume 1 provides an overview of the organizational structure and interfaces and the technical and administrative practices of the criticality protection policy and programs that are being developed for the SNF Project operations.

D6.6 CRITICALITY INSTRUMENTATION

This section addresses the need for a criticality alarm system and a criticality detection system in the ISA. DOE Order 5480.24 references ANSI/ANS-8.3-1997, *Criticality Accident Alarm System*, for requirements relating to nuclear criticality alarm systems. ANSI/ANS-8.3-1997 states that neither a criticality alarm system or criticality detection system is required, where the probability of a criticality accident is determined to be less than 1×10^{-6} per year. Interpretive guidance on the probability determination (Holten 1993) states: "The use of 10^{-6} does not necessarily mean that a PRA [probabilistic risk assessment] has to be performed. Reasonable grounds shall be presented on the basis of commonly accepted engineering judgment." Accordingly, the remaining discussion supports the judgment that no criticality alarm or detection systems are required in the ISA.

The hazard evaluation in Section D6.3.2 has identified credible hazardous conditions related to criticality at the ISA. The results of the criticality analyses (Section D6.3.3) show that with the engineered and administrative controls identified in Section D6.4.2, all credible hazardous conditions at the ISA are safely subcritical. Therefore, no criticality alarm system or criticality detection system is required at the ISA. The engineered and administrative controls in Section D6.4.2 are required to: (1) ensure that no unanalyzed hazardous configurations can exist at the ISA, and (2) protect the NAC-1 casks from hazardous conditions resulting from flooding of the casks.

D6.7 REFERENCES

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WHC-SD-FF-CSER-004, 1997b, *Criticality Safety Evaluation for the Long Term Storage of FFTF Fuel in Interim Storage Casks*, Rev. 1-B, Fluor Daniel Northwest, Richland, Washington.

WHC-SD-FF-CSER-006, 1995, *Criticality Safety Evaluation Report for TRIGA Fuel Storage at 400 Area Interim Storage Area,* Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC-SD-FV792-DA-004, 1991, *Criticality Safety Evaluation of Ident 69 in Core Component Container*, Rev. 0, EDT 109172, Westinghouse Hanford Company, Richland, Washington.

WHC-SD-NR-CSER-007, 1992, *Criticality Safety Evaluation Report, 105-KE Basin Fuel Encapsulation,* Rev. 1, Appendices A and B, Westinghouse Hanford Company, Richland, Washington.

WHC-SD-NR-SWD-024, *Software Certification Package for the GOLF Code*, Westinghouse Hanford Company, Richland, Washington.

WHC-SD-SQA-CSA-30006, 1995, *Criticality Safety Evaluation Report 95-012: Transfer of TRIGA Fuel from 308 Building Reactor Pool to Storage Casks,* Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC-SD-SQA-CSWD-20019, 1994, *MONK6A Pu Validation*, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC-SD-SQA-CSWD-20019, 1998, *Monk Validation Study Comment Resolution*, Rev. 0A, Westinghouse Hanford Company, Richland, Washington.

WHC-SD-TP-SARP-008, 2002, *Safety Analysis Report for Packaging NRF TRIGA Packaging,* Rev. 0-C, Fluor Hanford, Incorporated, Richland, Washington.

WHC-SD-TP-SARP-010, 1996, *Safety Analysis Report for Packaging (Onsite) Interim Storage Cask,* Rev. 0, Westinghouse Hanford Company, Richland, Washington.

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CHAPTER D7.0

RADIATION PROTECTION

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CONTENTS

LIST OF FIGURES

LIST OF TERMS

ALARA	as low as reasonably achievable
FFTF	Fast Flux Test Facility
ISA	interim storage area
ISC	interim storage cask
ISO	International Standards Organization
LWR	light water reactor
NRC	U.S. Nuclear Regulatory Commission
NRF	Neutron Radiography Facility
RCT	radiological control technician
SNF	spent nuclear fuel

D7.0 RADIATION PROTECTION

D7.1 INTRODUCTION

The essential features of the radiation protection programs that provide for radiation exposure control, radiological monitoring, and radiological protection instrumentation at all Spent Nuclear Fuel (SNF) Project facilities are addressed in Chapter 7.0 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1. Additional features of radiation protection specific to the 200 Area Interim Storage Area (ISA) are addressed in this Chapter D7.0. The primary function of the 200 Area ISA is to store non-defense reactor spent fuel housed in dry cask storage systems. The purpose of the 200 Area ISA is to consolidate the storage of three types of SNF at a new storage facility in the 200 Area. The fuel types are (1) Fast Flux Test Facility (FFTF) SNF, (2) Neutron Radiography Facility (NRF) TRIGA[1] SNF, and (3) 300 Area light water reactor (LWR) SNF. The main radiation protection features of the dry fuel storage systems include: (1) radiation shielding, (2) radioactive material confinement, and (3) minimizing external surface contamination.

A portion of the FFTF SNF is currently stored at the 400 Area ISA within a dry Interim Storage Cask (ISC). The NRF TRIGA SNF is also stored at the 400 Area ISA within a dry storage system consisting of either an NRF TRIGA cask or a DOT-6M[2] container that is housed within a right-circular cylinder concrete vault (Rad-Vault[3]). These storage systems have been analyzed, reviewed, and approved by the U.S. Department of Energy for storage of these types of fuel at FFTF. The results, although performed for the 400 Area ISA, can be used to assess functional requirements for the storage systems at the 200 Area ISA. The LWR SNF dry storage systems consist of a NAC-1 cask[4] within an International Standards Organization (ISO) container. The LWR storage system is not presently used for SNF storage at FFTF, but the system was previously analyzed with the intent to store the LWR fuel at the 400 Area ISA. The results of these analyzes provide information relative to the performance of the storage system that can be used to assess the functional requirements for the LWR SNF storage system at the 200 Area ISA.

The radiological protection features of these dry storage systems are sufficient to preclude onsite or offsite radiological doses in excess of the requirements established in Title 10, *Code of Federal Regulations*, Part 72, and Title 10, *Code of Federal Regulations*, Part 835, and ALARA (as low as reasonably achievable) goals. This is true for normal storage, off-normal conditions, and accident conditions. No exposures to airborne radioactive material are expected.

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[1] TRIGA is a trademark of General Dynamics Corporation.

[2] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.

[3] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.

[4] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

D7.2 REQUIREMENTS

The requirements that form the basis for the radiation protection program are identified in Section 7.2 of HNF-3553, Volume 1.

D7.3 RADIATION PROTECTION PROGRAM AND ORGANIZATION

The SNF Project radiation protection program and its organization, including safety management policies and philosophies, are described in Section 7.3 of HNF-3553, Volume 1.

D7.4 ALARA POLICY AND PROGRAM

A discussion of the SNF Project ALARA policy and program is provided in Section 7.4 of HNF-3553, Volume 1. The SNF Project policy regarding ALARA principles during 200 Area ISA design and construction is described in HNF-SD-SNF-PMP-018, *Site-Wide Spent Nuclear Fuel Project Management Plan.*

Detailed descriptions of the FFTF SNF dry storage system (ISC) shielding configuration are contained in Chapter D4.0. Based on the radiological protection features of the ISC, design analysis results in *FFTF Spent Fuel Interim Storage Cask Design Analysis Report* (General Atomics 1995) show that occupational exposure will be well below the ALARA requirements of 10 CFR 72, 10 CFR 835, and HNF-5173, *Project Hanford Radiological Control Manual.* The maximum exposure rates allowed by the design are 200 mrem/h at contact on the bottom surface and less than 2.0 mrem/h for all other surface areas. This higher level for the bottom surface is acceptable because it is normally not accessible. ISC dose rate-shielding analysis calculates the highest accessible surface dose rate to be 1.99 mrem/h at the top of the cask and 99.3 mrem/h at the bottom closure.

A detailed description of the TRIGA/Rad-Vault dry storage system shielding configuration is provided in Chapter D4.0. Based on the radiological protection features of the TRIGA/Rad-Vault, design analysis results reported in WHC-SD-FF-TI-043, *Safety Analysis Calculations for TRIGA Fuel Storage at 400 Area Interim Storage Area*, show the highest contact dose rate to be 0.76 mrem/h at the side of the concrete vault. The maximum exposure rate allowed by the design is <5.0 mrem/h at contact with the vault surface.

Detailed descriptions of the LWR SNF dry storage system shielding configuration are contained in Chapter D4.0. Based on the radiological protection features of the NAC-1 cask, design analysis results reported in HNF-3016, *NAC-1 Cask Dose Rate Calculations for LWR Spent Fuel*, show that occupational exposure will be well below the ALARA requirements of 10 CFR 72, 10 CFR 835, and HNF-5173. The NAC-1 dose rate-shielding analyses calculate the highest accessible surface dose to be less than 114 mrem/h. The maximum exposure rates allowed by the design are 200 mrem/h at contact on any surface or 10 mrem/h at 2 meters away from the external surface of the cask.

Routine and accidental releases to the public (offsite receptor) are calculated at the Site boundary shown in Figure D1-2. This is also the location of the controlled area boundary as the term is defined in 10 CFR 72.106. The shielding for all of the dry storage systems also meets the HNF-5173 requirement for uncontrolled access by the public of less than 0.05 mrem/h (at the ISA fence, see Figure D7-1).

Specific 200 Area ISA design features credited with reducing radiation exposure and achieving ALARA program objectives include the storage system shielding.

D7.5 RADIOLOGICAL PROTECTION TRAINING

SNF Project requirements and criteria for radiological protection training are described in Section 7.5 of HNF-3553, Volume 1.

D7.6 RADIATION EXPOSURE CONTROL

A description of SNF Project radiation exposure control measures is provided in Section 7.6 of HNF-3553, Volume 1. The SNF Project has elected to incorporate in the facility design the following U.S. Nuclear Regulatory Commission (NRC) requirements:

- Apply the radiological exposure criteria of 10 CFR 72.104, "Criteria for Radioactive Materials in Effluents and Direct Radiation from an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage," to the design and safety analyses.
- Apply the hourly dose limit of Title 10, *Code of Federal Regulations*, Part 20, "Standards for Protection Against Radiation," Section 20.1301, "Dose Limits for Individual Members of the Public," to the design and safety analyses.
- Incorporate control devices in the facility design for access to high-radiation areas that conform to the requirements of 10 CFR 20.1601, "Control of Access to High Radiation Areas."

For the 200 Area ISA, the radiological exposure annual dose criteria of 10 CFR 72.104 have been incorporated into the storage system designs and safety analysis. These criteria apply to design measures to protect any offsite public individual during normal operations and anticipated occurrences. These annual dose equivalent criteria are 25 mrem (0.25 mSv) to the whole body, 75 mrem (0.75 mSv) to the thyroid, and 25 mrem (0.25 mSv) to any other critical organ.

For the 200 Area ISA, the criteria for the hourly dose limit to the public, as described in 10 CFR 20.1301, have been incorporated into the design and safety analysis. This dose limit (0.002 rem) is assumed to be direct radiation from external sources (casks) for any unrestricted area during normal operations and anticipated occurrences.

During operation, access to the radiological area of the ISA (see Figure D7-1) is controlled using one or more of the methods listed and discussed in Section 7.6.2.6 of HNF-3553, Volume 1. The potential for the highest dose rates at the ISA occurs in the NAC-1 ISO containers. To avoid unnecessary and inadvertent doses to personnel, the doors to the ISO containers are provided with locks and security seals that can be used to control access. Primarily, access control will be by the use of signs and barricades and the locks on the ISA entry gates and the ISO container doors. Other measures, described in Section 7.6.2.6 of HNF-3553, Volume 1, will be used if needed.

D7.7 RADIOLOGICAL MONITORING

The radioactive material sampling and monitoring programs conducted within SNF Project facilities are addressed in Section 7.7 of HNF-3553, Volume 1.

D7.8 RADIOLOGICAL PROTECTION INSTRUMENTATION

A summary of the SNF Project plans and procedures governing radiation protection instrumentation is provided in Section 7.8 of HNF-3553, Volume 1. Only portable radiation protection instrumentation is used at the ISA.

D7.9 RADIOLOGICAL PROTECTION RECORD KEEPING

Radiological protection record-keeping requirements are described in Section 7.9 of HNF-3553, Volume 1.

D7.10 OCCUPATIONAL RADIATION EXPOSURES

For the 200 Area ISA, estimated radiation doses to onsite workers were calculated by assessing the operational procedures and planned activities that would result in occupational exposures. Using the estimated amount of time a worker would be near the storage system during handling, surveillance, and maintenance, the estimated occupational exposure to facility workers will be below the 500 mrem/yr administrative control level of HNF-5173.

Annual surveillance of the ISCs will include visually inspecting the casks and performing a radiological survey in the immediate area of the ISC. Operational analysis indicates that these activities will involve one operator and two radiological control technicians (RCTs), and will require about one hour to complete. Using worst-case dose rates, occupational exposure calculations indicate an annual accumulated dose of approximately 320 mrem to all workers during the inspection of fifty-three casks.

Annual surveillance of the TRIGA casks and the DOT-6M containers will include removing the Rad-Vault lid, visually inspecting the casks and containers, performing a radiological survey, and replacing the lid. Operational analysis indicates that these activities will involve one operator, two RCTs, and support personnel, and will require about two hours to complete. Radiological dose rate calculations for the NRF TRIGA casks were conservatively based on 19 maximally exposed fuel elements with an assumed prior irradiation, and an assumed 6-year decay time. (The decay time assumed was from the last powered operation of the TRIGA reactor in May 1989.) Actual cask readings that are representative of the maximum dose rate expected have shown 35 mrem/h (versus 81 mrem/h calculated) on the top and 50 mrem/h (versus 92 mrem/h calculated) on the side. It would naturally follow that the dose at the container could also be expected to be lower than previously calculated. Using worst-case dose rates, occupational exposure calculations indicate an annual accumulated dose of approximately 210 mrem to all workers during inspection of the vault. The exposure to crane operators removing and replacing the lid of the vault would be negligible.

Annual surveillance of the NAC-1 cask will include opening the ISO container door, visually inspecting the cask, and performing a radiological survey. Operational analysis indicates that these activities will involve two operators and one RCT, and will require about two hours to complete. Using worst-case dose rates, occupational exposure calculations indicate an annual accumulated dose of approximately 600 mrem to all workers during inspection of the six casks.

The maximum anticipated exposure resulting from these surveillances and other activities involving lesser exposures is 1150 mrem/year. The majority of this exposure is received by the RCTs. Using an RCT staffing level of four individuals involved in these annual inspections, the average RCT dose per year is 288 mrem total effective dose equivalent, which is well below the 500 mrem (5 mSv) ALARA design goal. The exposure level of all other categories of workers is considerably below this RCT exposure level.

Handling of the ISCs is anticipated to include lifting fixture connection and removal, ISC movement, and installation of the environmental cover. Operational analysis indicates that these activities will involve one rigger and one RCT, and will require about one hour per ISC. These activities primarily require personnel to be near the top of the cask where the exposure is the highest. Using worst-case dose rates, occupational exposure calculations indicate an accumulative dose of approximately 210 mrem to all workers during the handling of fifty-three casks.

Based on FFTF Rad-Vault loading experience, the estimated total personnel exposure time to load the Rad-Vault with two DOT-6M containers and six TRIGA casks is approximately 4 hrs. Operational analysis indicates that these activities will involve one operator and one RCT, and that the exposure levels experienced are similar to those involved with the Rad-Vault surveillance described above (i.e., each occurs with the Rad-Vault lid removed and with personnel above the open storage casks). Occupational exposure calculations indicate an accumulative dose of approximately 280 mrem to all workers during vault loading.

Handling of the NAC-1 casks is anticipated to primarily include inspection of the cask and ISO container after it is in place, since the connection and disconnection of the handling fixture is remotely operated. Operational analysis indicates that this activity will involve one operator and one RCT, and will take less than 15 minutes to complete. Using worst-case dose rates, occupational exposure calculations indicate an accumulative dose of approximately 300 mrem to all workers during handling of six casks.

The maximum anticipated exposure resulting from these handling and loading activities is 790 mrem/yr. Using a conservative staffing level of four individuals, the average worker dose per year is 197 mrem total effective dose equivalent, which is well below the 500 mrem (5 mSv) ALARA design goal. It is not anticipated that cask handling and vault loading will occur in the same year as system surveillances; therefore, only one of the anticipated annual exposures needs to be considered in evaluating compliance to the ALARA goal.

D7.11 REFERENCES

10 CFR 20, "Standards for Protection Against Radiation," *Code of Federal Regulations*.

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

10 CFR 835, "Occupational Radiation Protection," *Code of Federal Regulations*, as amended.

General Atomics, 1995, *FFTF Spent Fuel Interim Storage Cask Design Analysis Report*, Document No. 910683, Rev. A, General Atomics Company, Incorporated, San Diego, California.

HNF-3016, 1998, *NAC-1 Cask Dose Rate Calculations for LWR Spent Fuel*, Rev. 0, DE&S Hanford, Incorporated, Richland, Washington.

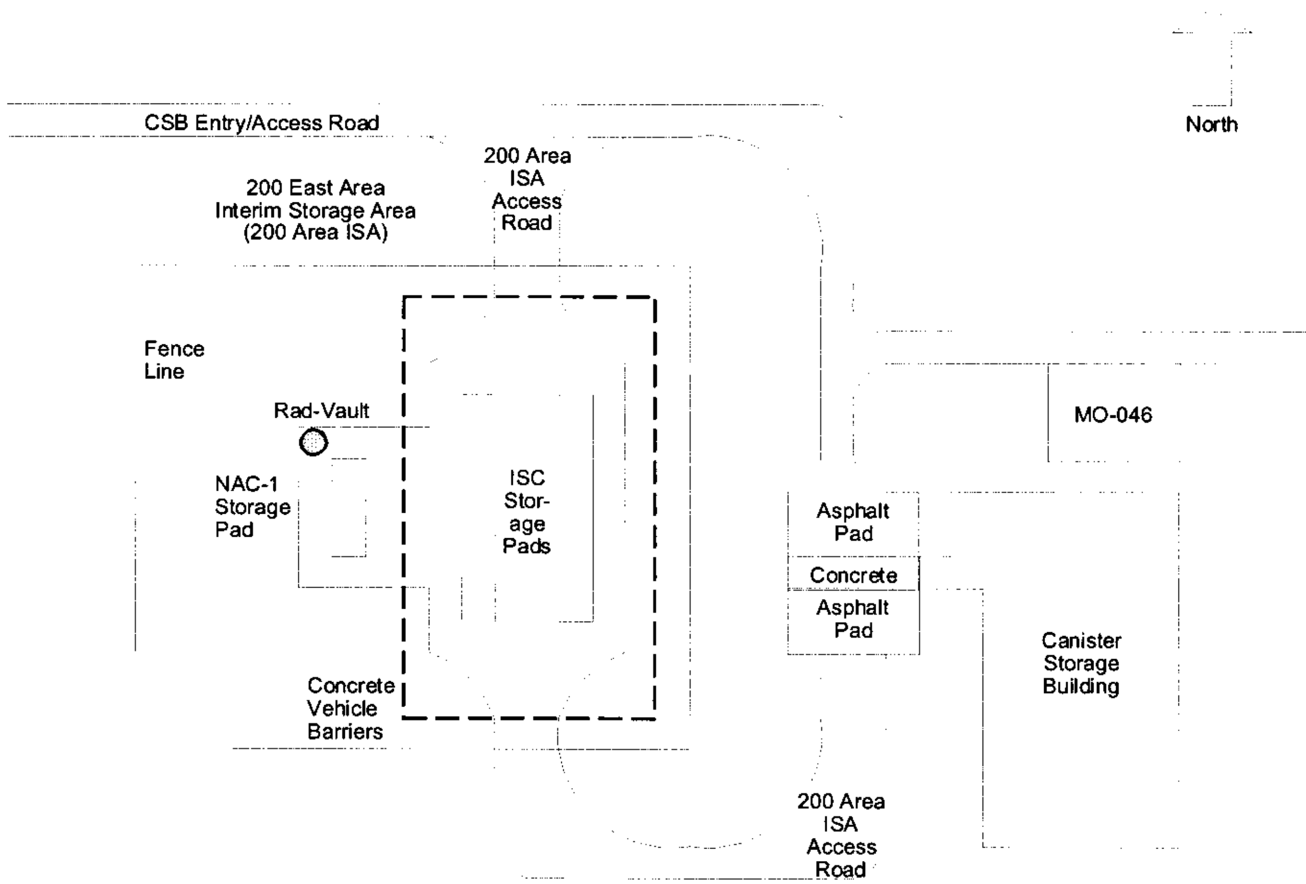
HNF-3553, 2000, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Vol. 1, Rev. 0-B, Fluor Hanford, Incorporated, Richland, Washington.

HNF-5173, 2001, *Project Hanford Radiological Control Manual*, Rev. 1, Fluor Hanford, Incorporated, Richland, Washington.

HNF-SD-SNF-PMP-018, 1997, *Site-Wide Spent Nuclear Fuel Project Management Plan*, Rev. 0, DE&S Hanford, Incorporated, Richland, Washington.

WHC-SD-FF-TI-043, 1995, *Safety Analysis Calculations for TRIGA Fuel Storage at 400 Area Interim Storage Area*, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

Figure D7-1. Radiological Control Boundary.



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CHAPTER D8.0

HAZARDOUS MATERIAL PROTECTION

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CONTENTS

LIST OF TERMS

ALARA	as low as reasonably achievable
ISA	interim storage area
ISC	interim storage cask
ISO	International Standards Organization
SNF	spent nuclear fuel

D8.0 HAZARDOUS MATERIAL PROTECTION

D8.1 INTRODUCTION

The major provisions of the occupational safety and health program, as the program applies to hazardous material protection for the Spent Nuclear Fuel (SNF) Project, are addressed in Chapter 8.0 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1.

D8.2 REQUIREMENTS

The requirements that form the basis for the hazardous material protection program are identified in Section 8.2 of HNF-3553, Volume 1.

D8.3 HAZARDOUS MATERIAL PROTECTION PROGRAM AND ORGANIZATION

The SNF Project has established a visible and comprehensive occupational safety and health program. This program is described in Section 8.3 of HNF-3553, Volume 1.

D8.4 ALARA POLICY AND PROGRAMS

While there is no established formal SNF Project ALARA (as low as reasonably achievable) program for nonradiological hazardous materials, the SNF Project has expanded the classic concept of ALARA (i.e., minimization of radiological exposures) to the application of exposure minimization for hazardous substances and conditions. The SNF Project's policy is described in Section 8.4 of HNF-3553, Volume 1.

Work practices for hazardous material protection and control of chemical exposures, as stated in Section 8.4 of HNF-3553, Volume 1, will be implemented at the 200 Area Interim Storage Area (ISA) using approved SNF Project implementing procedures. The occupational safety and health program will use the additional provisions of Section 8.4 of HNF-3553, Volume 1. These provisions will also be implemented at the ISA using approved SNF Project implementing procedures.

Applicable ergonomics considerations based on DOE Order 5480.10, *Contractor Industrial Hygiene Program* (Section 9B), under the occupational safety and health program, industrial hygiene subprogram, are included in Chapter D13.0. Ergonomics considerations, along with risk factor control processes and the ergonomics program, are described in the SNF Project human engineering program plan (see Section 8.4 of HNF-3553, Volume 1, and SNF-4399, *Human Engineering Program Plan*). This document will be revised as necessary to address integration of this plan with HNF-MP-003, *Integrated Environment, Safety, and Health Management System Description*.

D8.5 HAZARDOUS MATERIAL TRAINING

Plans and procedures for training SNF Project workers regarding hazardous materials are summarized in Section 8.5 of HNF-3553, Volume 1.

Section 8.5 of HNF-3553, Volume 1, states that SNF Project management provides training, professional education, and certification opportunities necessary to support, maintain, and enhance industrial hygiene staff proficiency to meet or exceed U.S. Department of Energy industrial hygiene training objectives and goals in accordance with HNF-SD-SNF-RD-001, *Spent Nuclear Fuel Project Standards/Requirements Identification Document*. ISA management is responsible for ensuring that workers assigned to any task involving hazardous materials are trained in the safety and health hazards associated with such hazardous materials. Workers will perform only those tasks for which they have received the proper training. If the ISA mission changes, ISA management will review the training requirements and modify them accordingly. ISA management is also responsible for ensuring that retraining is provided within the time allowed by training course requirements.

D8.6 HAZARDOUS MATERIAL EXPOSURE CONTROL

Worker safety features at the ISA are an integral part of facility design and operation. The ISA design encompasses human factors considerations to ensure that operations can be conducted safely. SNF Project occupational exposures to hazardous materials and the spread of hazardous material contamination are controlled by a combination of engineered, operational, and administrative controls, and by the use of personal protective clothing and equipment. These controls are described in Section 8.6 of HNF-3553, Volume 1.

Construction of the ISA has been performed to minimize the use of hazardous materials and the generation of hazardous and non-hazardous waste. No polychlorinated biphenyls or asbestos will be used. Transformers, lighting, and other electrical equipment that use an insulating oil will be polychlorinated biphenyl-free.

No significant hazardous materials have been identified for the ISA as a result of a hazard analysis that was performed and documented in SNF-4820, *200 Area Interim Storage Area Final Hazard Analysis Report*, except for the radionuclide content in the dry cask storage systems. HNF-2524, *200 East Area Interim Storage Area Preliminary Safety Evaluation*, states that these materials are primarily uranium oxide and mixed uranium and plutonium oxide, which are known to have toxicological effects. As stated in Section 3.1.1 of HNF-2524, the toxicological hazards of the radionuclide inventory were found to be bounded by the radiological consequences.

Small quantities of other hazardous material identified by the hazard identification process include pyrophoric metals and hydrides, oxidizers, hydrogen, diesel fuel, and other flammable or combustible materials. No routine chemical processes will be conducted in the ISA. Some chemicals, such as those used for equipment decontamination, may be used occasionally. Table D3-1 identifies hazards by form, type, location, and total quantity.

The FFTF SNF inventory includes a small amount of metallic sodium inside the sodium-bonded fuel pins. Purging and backfilling of pressure monitoring equipment will involve the use of an inert gas. In addition, some chemicals, such as those used for equipment decontamination, may be used occasionally.

An inert atmosphere (helium and argon) is maintained in storage containers for FFTF SNF. This inherently precludes a potential hazard of pyrophoric uranium or plutonium reacting with air (oxygen) that may have leaked in the SNF container. In addition, there is a potential for the sodium to react with air (oxygen). Potential for this event is low since only 8 of 329 FFTF items are metallic fuel; the remainder are not pyrophoric. In addition, the inert atmosphere will preclude reaction of air with either the metallic uranium or sodium. These eight items will be consolidated into two casks.

Major features of worker protection are presented in Table D3-4 and are categorized by hazard. These features are in addition to safety-class or safety-significant features for design basis accidents. No safety-significant structures, systems, and components or Technical Safety Requirements have been identified for the ISA based solely on worker safety considerations.

All work activities in the ISA will receive adequate advance planning so that if potential hazardous materials are identified due to changing conditions in the future, specific precautions will be applied. The exposure controls identified in Sections 8.6.1 through 8.6.4 of HNF-3553, Volume 1, will then be implemented at the ISA using approved SNF Project implementing procedures.

Other hazardous materials at the ISA (maintenance materials) will be properly inventoried and stored to control hazards inherent to the material, in accordance with SNF Project implementing procedures.

D8.7 HAZARDOUS MATERIAL MONITORING

Summaries of the hazardous material sampling and monitoring programs that are conducted internally and externally for SNF Project facilities are provided in Section 8.7 of HNF-3553, Volume 1. The workplace and external monitoring program described in Sections 8.7.1 and 8.7.2 of HNF-3553, Volume 1, will be implemented at the ISA, as appropriate, using approved SNF Project implementing procedures. An environmental, radioactivity, and chemical emissions monitoring program, including requirements, is presented in Section 8.7.2 of HNF-3553, Volume 1. This program will be implemented for the ISA in accordance with approved SNF Project implementing procedures.

Surveillance and monitoring activities to ensure safe storage of SNF at the ISA are presented in Section D2.3.2. These activities include the following:

- Annual surveillance of the interim storage casks (ISC) – This surveillance involves a visual inspection, radiation survey and smear sampling about the ISC environmental cover.

- Annual surveillance of TRIGA[1] fuels in DOT-6M[2] containers – This surveillance includes a visual inspection and radiation surveys of the Rad-Vault,[3] visual inspection and radiation survey of the fuel cask, and smear sampling of the fuel casks.

- Annual surveillance of the NAC-1 casks[4] in International Standards Organization (ISO) containers – The surveillance includes a visual inspection and radiation survey of the ISO container external surface, radiation survey and smear samples of the top of NAC-1 casks, and visual inspection of the interior of the ISO container and exterior of the NAC-1 casks.

D8.8 HAZARDOUS MATERIAL PROTECTION INSTRUMENTATION

Summaries of plans and procedures governing hazardous protection instrumentation are provided in Section 8.8 and Table 8-2 of HNF-3553, Volume 1. As stated in Section 8.8 of HNF-3553, Volume 1, safety and health specialists will determine the need for hazardous protection instrumentation and the number and placement of instruments under normal and emergency conditions, in accordance with the requirements stated in Section 8.8 of HNF-3553, Volume 1.

D8.9 HAZARDOUS MATERIAL PROTECTION RECORD KEEPING

The SNF Project has an established document control and records management program. This program is summarized in Section 8.9 of HNF-3553, Volume 1.

D8.10 HAZARD COMMUNICATION PROGRAM

The hazard communication program applies to the purchase, receipt, transportation, use, and storage of hazardous chemicals and products. This program is summarized in Section 8.10 of HNF-3553, Volume 1. The hazard communication program for the ISA will be implemented in accordance with the provisions of Sections 8.10.1 through 8.10.6 of HNF-3553, Volume 1, including hazard posting in work areas, chemical management, chemical labeling, chemical

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[1] TRIGA is a trademark of General Dynamics Corporation.

[2] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.

[3] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.

[4] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

product list, material safety data sheets, information, and training. This program will be accomplished in accordance with approved SNF Project implementing procedures.

D8.11 OCCUPATIONAL CHEMICAL EXPOSURES

Predicted annual exposures to workers from hazardous material sources are identified in Section 8.11 of HNF-3553, Volume 1. The identification of chemical hazard locations, posting, chemical management, and other controls to limit occupational chemical exposure is included in Section 8.10 of HNF-3553, Volume 1.

D8.12 REFERENCES

DOE Order 5480.10, 1985, *Contractor Industrial Hygiene Program*, U.S. Department of Energy, Washington, D.C.

HNF-2524, 1998, *200 East Area Interim Storage Area Preliminary Safety Evaluation*, Rev. 0, DE&S Hanford, Incorporated, Richland, Washington.

HNF-3553, 2000, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Vol. 1, Rev. 0-B, Fluor Hanford, Incorporated, Richland, Washington.

HNF-MP-003, 2001, *Integrated Environment, Safety, and Health Management System Description*, Rev. 4, Fluor Hanford, Incorporated, Richland, Washington.

HNF-SD-SNF-RD-001, 2000, *Spent Nuclear Fuel Project Standards/Requirements Identification Document*, Rev. 3, Fluor Hanford, Incorporated, Richland, Washington.

SNF-4399, 1999, *Human Engineering Program Plan*, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

SNF-4820, 1999, *200 Area Interim Storage Area Final Hazard Analysis Report*, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

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CHAPTER D9.0

RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT

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CONTENTS

LIST OF TERMS

ISA	interim storage area
SNF	spent nuclear fuel

D9.0 RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT

D9.1 INTRODUCTION

The essential features of the radioactive and hazardous waste management programs that provide for the safe control, collection, and handling of wastes generated during routine operations at Spent Nuclear Fuel (SNF) Project facilities are detailed in Chapter 9.0 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1. This Chapter D9.0 only applies to waste generated within the 200 Area Interim Storage Area (ISA) and those systems designed to deal with that waste. The 200 Area ISA will store SNF in dry cask storage systems. No uncontained radioactive materials will be handled. Spent fuel in shipping containers is received, unloaded, and stored at the 200 Area ISA. No chemical, toxicological, or hazardous materials will normally exist onsite except for the SNF within the storage systems. Confinement of the 200 Area ISA radioactive materials is a design feature of each dry spent fuel storage system.

D9.2 REQUIREMENTS

The requirements that form the basis for the radioactive and hazardous waste management program are found in Section 9.2 of HNF-3553, Volume 1.

D9.3 RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT PROGRAM AND ORGANIZATION

The facility administrative procedures for solid waste management contain the procedural guidance for the planning, generation, and disposal of generated waste in compliance with applicable requirements. The administrative procedures cover characterization, preplanning, designation, containerization, disposal, and programmatic requirements. A summary of the SNF Project waste management program is provided in Section 9.3 of HNF-3553, Volume 1.

D9.4 RADIOACTIVE AND HAZARDOUS WASTE STREAMS AND SOURCES

The only waste streams at the ISA are solid wastes; primarily light bulbs, vegetation and animal carcasses, and waste generated during contamination monitoring. The estimated volumes for these wastes are:

- Sodium light bulbs - < 1 L/yr (0.25 gal/yr) (recycled)
- Miscellaneous solid waste (primarily rags, wipes, protective clothing, vegetation, and carcasses) - < 1 m^3/yr (35 ft^3/yr) (potentially radioactive).

In the unlikely event of a radioactive release, cleanup materials will be designated as low-level waste. Since the 200 Area ISA is not manned, no sanitary sewage is generated. The low-level waste will be packaged per the Waste Management Technical Services waste acceptance requirements in HNF-EP-0063, *Hanford Site Solid Waste Acceptance Criteria* (described in Section 16 of HNF-SD-SNF-RD-001, *Spent Nuclear Fuel Project Standards/Requirements Identification Document,* and in facility waste administrative procedures) and then transported to Waste Management Technical Services for disposition. Sodium light bulbs will be delivered to the 400 Area consolidation area where they will be picked up by commercial treatment and disposal facility operators. Miscellaneous nonradioactive waste will be disposed of at an offsite waste disposal site.

D9.5 REFERENCES

HNF-3553, 2000, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Vol. 1, Rev. 0-B, Fluor Hanford, Incorporated, Richland, Washington.

HNF-EP-0063, 2001, *Hanford Site Solid Waste Acceptance Criteria*, Rev. 6, Fluor Hanford, Incorporated, Richland, Washington.

HNF-SD-SNF-RD-001, 2000, *Spent Nuclear Fuel Project Standards/Requirements Identification Document*, Rev. 3, Fluor Hanford, Incorporated, Richland, Washington.

CHAPTER D10.0

INITIAL TESTING, IN-SERVICE SURVEILLANCE, AND MAINTENANCE

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CONTENTS

LIST OF TERMS

DOE	U.S. Department of Energy
ISA	interim storage area
ISC	interim storage cask
ISO	International Standards Organization
SNF	spent nuclear fuel

D10.0 INITIAL TESTING, IN-SERVICE SURVEILLANCE, AND MAINTENANCE

D10.1 INTRODUCTION

Essential features of the initial testing program, operational readiness review, in-service surveillance program, and the maintenance program implemented at Spent Nuclear Fuel (SNF) Project facilities are described in Chapter 10.0 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1. 200 Area Interim Storage Area (ISA)-specific features of these programs are described in this chapter.

D10.2 REQUIREMENTS

The requirements that form the basis for the initial testing, surveillance, and maintenance programs are found in Section 10.2 of HNF-3553, Volume 1. Specific requirements applicable to this chapter include the following:

- U.S. Department of Energy (DOE) O 430.1A, *Life Cycle Asset Management*
- DOE Order 4330.4B, *Maintenance Management Program*
- DOE Order 5480.19, *Conduct of Operations Requirements for DOE Facilities*.

D10.3 INITIAL TESTING

The SNF Project initial testing program ensures the operability of equipment and facilities before facility operation. Project details of this program are provided in Section 10.3 of HNF-3553, Volume 1. Due to the simple passive nature of the facility and its components, the 200 Area ISA initial testing program will consist entirely of construction acceptance testing. No special testing requirements prior to receipt of SNF have been identified for the 200 Area ISA systems and components.

D10.4 IN-SERVICE SURVEILLANCE PROGRAM

The SNF Project in-service surveillance program is designed to maintain the integrity of facility systems and to ensure that systems perform their function of protecting the health and safety of the public, workers, and facility staff by preventing or mitigating accident consequences. Details of this program are provided in Section 10.4 of HNF-3553, Volume 1. The Technical Safety Requirements described in Chapter D5.0 identify no safety surveillance requirements for the 200 Area ISA.

Periodic inspection of the interim storage casks (ISCs) is required to support analysis assumptions. This annual surveillance will include a visual inspection, radiation survey, and smear sampling about the ISC environmental cover.

Annual surveillance of the TRIGA[1] cask and DOT-6M[2] containers will include (1) visual inspection and radiation survey of the Rad-Vault,[3] (2) vault lid removal to allow visual inspection and radiation surveys of the fuel casks/containers, and (3) smear sampling of the fuel casks/containers. Since the Rad-Vault is constructed of concrete, an annual visual inspection of the underside is not required.

HNF-10094, *NRF TRIGA Cask Seal Plan* establishes a seal integrity program for TRIGA fuel storage and related preventive maintenance and testing requirements, and identifies requirements for TRIGA fuel containers to be accomplished prior to year 2014 to protect seal design life considerations.

Annual surveillance of the NAC-1 casks[4] in International Standards Organization (ISO) containers will include (1) a visual inspection of the external surface of the ISO container, (2) radiation surveys of the exterior surface of the ISO, (3) a radiation survey of the top of the NAC-1 casks upon opening the ISO container doors, (4) smear samples from the top of the NAC-1 casks, and (5) visual inspections of the interior of the ISO container and exteriors of the NAC-1 casks. The ISO has structural members constructed of ferrous materials, and an overall inspection including the underside is appropriate every 5 years.

D10.5 MAINTENANCE PROGRAM

The maintenance program for the SNF Project facilities is conducted in accordance with DOE Order 4330.4B, which provides the general policy and objectives for establishing cost-effective maintenance and repair programs for DOE property. The maintenance program will incorporate the results of considerable project and subsystem vendor interface activities aimed at ensuring an acceptable design and acceptable operating practices relative to reliability, availability, and maintainability.

Policies and procedures are in place to effectively manage SNF Project facility maintenance activities. Section 10.5 of HNF-3553, Volume 1, summarizes the maintenance policies and procedures that are implemented at the 200 Area ISA. The 200 Area ISA facility components are classified as General Service. The dry cask systems have been designated Safety Significant. Due to the simple passive nature of the facility and its components, minimal maintenance activities have been identified by the SNF Project for this facility. Identified maintenance tasks include monitoring and maintaining the equipment for receipt of SNF storage systems. During long-term storage, maintenance tasks include painting of the storage cask systems, lamping, fence and gate inspections and repairs, and vegetation removal. The Viton O-rings on the Neutron Radiography Facility TRIGA cask have a 20-year design life (per vendor) and require evaluation prior to year 2015.

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[1] TRIGA is a trademark of General Dynamics Corporation.

[2] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.

[3] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.

[4] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

D10.6 REFERENCES

DOE O 430.1A, 1998, *Life Cycle Asset Management*, U.S. Department of Energy, Washington, D.C.

DOE Order 4330.4B, 1994, *Maintenance Management Program*, U.S. Department of Energy, Washington, D.C.

DOE Order 5480.19, 1992, *Conduct of Operations Requirements for DOE Facilities*, U.S. Department of Energy, Washington, D.C.

HNF-3553, 2000, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Vol. 1, Rev. 0-B, Fluor Hanford, Incorporated, Richland, Washington.

HNF-10094, 2002, *NRF TRIGA Cask Seal Plan*, Rev. 0, Fluor Hanford, Incorporated, Richland, Washington.

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CHAPTER D11.0

OPERATIONAL SAFETY

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CONTENTS

LIST OF TERMS

CFR	Code of Federal Regulations
DOE	U.S. Department of Energy
FFTF	Fast Flux Test Facility
FHA	fire hazards analysis
ISA	interim storage area
ISC	interim storage cask
ISO	International Standards Organization
MPFL	maximum possible fire loss
NFPA	National Fire Protection Association
NRF	Neutron Radiography Facility
SNF	spent nuclear fuel

D11.0 OPERATIONAL SAFETY

D11.1 INTRODUCTION

Features of the Spent Nuclear Fuel (SNF) Project conduct of operations and fire protection programs are described in the following sections and in Chapter 11.0 of HNF-3553, *Spent Nuclear Project Final Safety Analysis Report*, Volume 1.

D11.2 REQUIREMENTS

The requirements that establish the basis for conduct of operations and general aspects of operational safety are identified in Section 11.2 of HNF-3553, Volume 1. Specific requirements applicable to this chapter include the following:

- DOE Order 5480.7A, *Fire Protection*
- RLID 5480.7, *Fire Protection.*

D11.3 CONDUCT OF OPERATIONS

"Conduct of operations" is a set of principles that establishes an overall philosophy for achieving excellence in the operation of the SNF Project facilities. SNF Project application of conduct of operations principles is described in Section 11.3 of HNF-3553, Volume 1.

D11.4 FIRE PROTECTION

The fundamental fire protection programs for SNF Project facilities are addressed in Section 11.4 of HNF-3553, Volume 1. The elements of the fire protection program that are specific to the 200 Area Interim Storage Area (ISA) are described in the following subsections. The results of a 200 Area ISA fire hazards analysis (FHA), documented in SNF-4932, *Fire Hazards Analysis for the 200 Area Interim Storage Area*, are addressed in Section D3.3. In addition, an FHA was performed for the proposed ISA storage building and the results are documented in HNF-4552, *Preliminary Fire Hazards Analysis for the 200 East Area Interim Storage Area Storage Building.* Results from these analyses are summarized in the following subsections.

D11.4.1 Fire Hazards

The FHAs (SNF-4932, HNF-4552) comprehensively assessed the risk from fire at the 200 Area ISA and storage building to determine that: (1) the potential for occurrence of fire is minimized, (2) a fire would not cause an onsite or offsite release of radiological and other hazardous materials that would threaten the public health and safety or the environment,

(3) requirements are in place that will provide an acceptable degree of life safety to SNF Project and contractor workers, and (4) the safety systems are not damaged by fire.

SNF-4932 and HNF-4552 were prepared to meet the requirements of U.S. Department of Energy (DOE) Order 5480.7A and to evaluate compliance to DOE fire protection criteria. As required, this analysis addressed the following elements:

- Description of construction
- Protection of essential safety-class equipment
- Fire protection features
- Description of fire hazards
- Life safety considerations
- Critical process equipment
- High value property
- Damage potential; maximum credible fire loss and maximum possible fire loss
- Fire department or brigade response
- Recovery potential
- Potential for toxic, biological, and/or radiological incident due to a fire
- Emergency planning
- Security considerations related to fire protection
- Natural hazards (earthquake, flood, wind) impact on fire safety
- Exposure fire potential, including the potential for fire spread between fire areas.

Future changes to the FHAs will be screened to these elements to ensure that no unreviewed safety question is created.

The ISA facility FHA (SNF-4932) identifies no fire hazards inherent to the 200 Area ISA components, which include interim storage casks (ISCs), Neutron Radiography Facility (NRF) TRIGA[1] casks, DOT-6M[2] containers within a Rad-Vault[3] storage vault, and NAC-1 casks[4] within International Standards Organization (ISO) containers. Potential fire hazards identified in the FHA include (1) vehicles and equipment required to move the cask storage systems (fuel truck precluded), and (2) combustible materials required during the cask storage system movement activities. Since no credible fire results in damage to the cask storage systems, the maximum possible fire loss (MPFL) is identified as the cost to inspect and evaluate the cask storage systems following exposure to fire. The MPFL value is $100,000. No release of radionuclides to the environment from the ISA facility is identified in the FHA, and no toxicological or biological consequences resulting from fire are anticipated. Additional information is provided in the FHA (SNF-4932).

The proposed ISA storage building will be used to store the excessed Fast Flux Test Facility (FFTF) solid waste transfer cask and ISA equipment (e.g., lifting devices, impact

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[1] TRIGA is a trademark of General Dynamics Corporation.
[2] DOT-6M containers are manufactured to the standards of the U.S. Department of Transportation.
[3] Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.
[4] NAC-1 casks are manufactured by Nuclear Assurance Corporation.

limiters, and the tractor-trailer unit). The ISA storage building FHA (HNF-4552) identifies two hazards associated with this building: (1) a fire starting in the tractor-trailer and spreading throughout the facility, and (2) a fire starting in stored combustibles and spreading throughout the facility. The MPFL is identified as the complete loss of the building, with the exception of the FFTF cask. The MPFL value is $3.4 million, including property and cleanup losses. The FFTF cask, which is a one-of-a-kind, high-value property, would have limited damage and could be reused following inspection and evaluation. No release of radionuclides to the environment from the ISA storage building is identified in the FHA as a result of the MPFL event because the cask internals are the only planned component to contain radioactive material. No toxicological or biological consequences resulting from fire are anticipated. Additional information is provided in the FHA (HNF-4552).

D11.4.2 Fire Protection Program and Organization

The fire protection program for SNF Project facilities is structured and implemented in accordance with the operating contractor's safety management policies, philosophies, and criteria described in Section 11.4.2 of HNF-3553, Volume 1. ISA-specific aspects of the fire protection program are described in the following paragraphs.

The ISCs, NAC-1 casks, Rad-Vault, and NRF TRIGA casks have been analyzed to meet the radioactive material release criteria of Title 10, *Code of Federal Regulations*, Part 71, "Packaging and Transportation of Radioactive Material" (10 CFR 71), Section 71.51(a)(2) from exposure to the 10 CFR 71.73(c)(3) fire condition. The FHA for the ISA determined that accidents associated with the ISA are bounded by the transportation fire scenario, which would bound the tractor-trailer fire, mobile crane fire, and runaway fuel truck fire. Therefore, no ISA facility fire protection system is required. Additional information is provided in the FHA (SNF-4932).

One fire hydrant is located 150 ft southwest from the NAC-1/ISO pad (just outside the fence line), and a fire main connection is included in the ISA storage building (just outside the northwest fence line intersection). A pull box is provided at the ISA storage building to facilitate reporting of fires.

Automatic sprinkler protection is provided throughout the ISA storage building and is supplied by a connection to the 200 Area raw water system. The fire protection system is classified as General Service. A fire alarm system is provided that provides for transmission of signals to the Hanford Fire Department and to local building fire alarm annunciators. Portable fire extinguishers will also be available in the ISA storage building.

Although the proposed storage building will not be occupied on a full time basis, adequate life safety features are provided (emergency exit doors and illumination) as required by National Fire Protection Association (NFPA) 101, *Life Safety Code*. A perimeter gate at the ISA will remain unlocked while personnel are working within the ISA. Based on the open egress point and the absence of significant combustibles, the FHA for the ISA identified no life safety concerns.

D11.4.3 Combustible Loading Control

Ordinary combustibles are expected at the ISA facility. The area will be kept free of debris and vegetation, and the perimeter fence will keep transient debris away from the cask storage systems. Combustibles anticipated in the FHA include plywood cribbing, minimal quantities of combustibles to support maintenance activities (fuel truck precluded), and a minimal amount of local vegetation. SNF-4932 found that a combination of plywood and fuel oil was bounding to the scenarios evaluated and describes limits on the allowable quantity of plywood cribbing. Ordinary combustibles are expected in the ISA storage building; however, it will not be a general storage structure. Section 11.4.3 of HNF-3553, Volume 1, summarizes the SNF Project program used to prevent unnecessary combustible loadings in project facilities. The transient combustibles control program will be implemented through operating procedures that address the ISA limits established in Technical Safety Requirement AC5.12 (see Section D5.5.3.6).

D11.4.4 Fire Fighting Capabilities

The Hanford Fire Department maintains a training program for fire fighting, fire system testing and maintenance, and facility inspections. Fire-fighting capabilities that apply to all SNF Project facilities are addressed in Section 11.4.4 of HNF-3553, Volume 1. 200 Area ISA-specific fire response procedures are addressed in the following paragraphs.

Due to the absence of any fire detection and alarm system in the 200 Area ISA outside the storage building, personnel action is required to notify the Hanford Fire Department in the event of a fire. No brigade is planned for the ISA and storage building facilities. The standard response to an alarm condition in the 200 Area is by the Hanford Fire Department from the 200 Area fire station. Hanford Fire Department response time from the 200 Area fire station is approximately five minutes. This is the response time and the responder location assumed in the FHAs. Vehicle access is provided by a paved access road and compacted gravel. The Hanford Fire Department is fully staffed, trained, and equipped for emergency response.

SNF Project personnel are trained on the expected actions to be taken in case of a fire. Personnel are to notify the Hanford Fire Department, evacuate the facility, and follow approved fire response plans specific to the facility.

D11.4.5 Fire Fighting Readiness Assurance

A pre-fire plan for the ISA will be prepared by the Hanford Fire Department prior to facility operations. A summary of SNF Project fire prevention inspections, fire safety drills and exercises, and program record-keeping requirements is provided in Section 11.4.5 of HNF-3553, Volume 1.

D11.5 REFERENCES

10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations.*

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

HNF-3553, 2000, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Vol. 1, Rev. 0-B, Fluor Hanford, Incorporated, Richland, Washington.

HNF-4552, 1999, *Preliminary Fire Hazards Analysis for the 200 East Area Interim Storage Area Storage Building,* Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

NFPA 101, 1997, *Life Safety Code*, National Fire Protection Association, Quincy, Massachusetts.

RLID 5480.7, *Fire Protection*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

SNF-4932, 2000, *Fire Hazards Analysis for 200 Area Interim Storage Area*, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

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CHAPTER D12.0

PROCEDURES AND TRAINING

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CONTENTS

LIST OF TERMS

SNF	spent nuclear fuel

D12.0 PROCEDURES AND TRAINING

D12.1 INTRODUCTION

A description of the essential features of Spent Nuclear Fuel (SNF) Project procedures and training is provided in Chapter 12.0 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1.

D12.2 REQUIREMENTS

The requirements that form the basis for the SNF Project training and procedures programs are identified in Section 12.2 of HNF-3553, Volume 1.

D12.3 PROCEDURE PROGRAM

SNF Project activities are conducted in accordance with written procedures. A summary of the facility procedures program, including development and maintenance of procedures, is provided in Section 12.3 and its subsections in HNF-3553, Volume 1.

D12.4 TRAINING PROGRAM

The objective of the SNF Project personnel training program is to provide and maintain a qualified work force for safe and efficient facility operations. A summary of the SNF Project personnel training program—including training development, maintenance of training, and modification of training materials—is provided in Section 12.4 and its subsections in HNF-3553, Volume 1.

D12.5 REFERENCES

HNF-3553, 2000, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Vol. 1, Rev. 0-B, Fluor Hanford, Incorporated, Richland, Washington.

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CHAPTER D13.0

HUMAN FACTORS

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CONTENTS

LIST OF TERMS

DOE	U.S. Department of Energy
ISA	interim storage area
SSC	structure, system, and component

D13.0 HUMAN FACTORS

D13.1 INTRODUCTION

DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, allows for a graded approach to the application of human factors analysis to nuclear facilities. Per DOE-STD-3009-94, final safety analysis report discussions are to pertain only to the human-machine interfaces with safety structures, systems, and components (SSCs) and in proportion to the importance of those human-machine interfaces to the performance of the safety SSCs. Since the safety-class SSCs at the 200 Area Interim Storage Area (ISA) are all passive devices, there are no human-machine interfaces except during storage system unloading. Events associated with unloading have been analyzed in Chapter D3.0.

All equipment used for storage system handling at the ISA exists either at other U.S. Department of Energy (DOE) facilities or on the Hanford Site. The applicable procedures, training, and staffing requirements needed for 200 Area ISA operations and the existing equipment were previously defined and are currently being used at the 400 Area ISA (Fast Flux Test Facility). The storage system handling procedures and training requirements are in compliance with the DOE/RL-92-36, *Hanford Site Hoisting and Rigging Manual*. DOE/RL-92-36 is updated based on lessons learned both at the Hanford Site and at other DOE facilities. ISA surveillance and maintenance procedures will be prepared using the Spent Nuclear Fuel Project Procedures Writer's Guide, which encompasses the human factors requirements of DOE-STD-1029-92, *Writer's Guide for Technical Procedures*. A review of the detailed operations associated with storage system handling identified no hazards that warrant analysis by a human factors expert. The approach used to address human factors is deemed adequate based on the guidance provided in DOE-STD-3009-94.

D13.2 REQUIREMENTS

The requirements that establish the basis for human factors engineering and storage system handling are identified in the following:

- Title 10, *Code of Federal Regulations*, Part 830, "Nuclear Safety Management"
- DOE Order 5480.23, *Nuclear Safety Analysis Reports*
- DOE/RL-92-36, *Hanford Site Hoisting and Rigging Manual*.

D13.3 HUMAN FACTORS PROCESS

The human factors process is discussed in Section D13.1.

D13.4 IDENTIFICATION OF HUMAN-MACHINE INTERFACES

The safety-class SSCs at the 200 Area ISA are all passive devices; therefore, there are no human-machine interfaces except during storage system unloading. Events associated with unloading have been analyzed in Chapter D3.0. Critical lift procedures for cask handling activities will incorporate controls for the lifting restrictions identified in Chapter D5.0.

D13.5 OPTIMIZATION OF HUMAN-MACHINE INTERFACES

See Sections D13.1 and D13.4.

D13.6 REFERENCES

10 CFR 830, "Nuclear Safety Management," *Code of Federal Regulations*, as amended.

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

DOE/RL-92-36, 1993, *Hanford Site Hoisting and Rigging Manual*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

DOE-STD-1029-92, 1992, *Writer's Guide for Technical Procedures*, 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.

CHAPTER D14.0

QUALITY ASSURANCE

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CONTENTS

LIST OF TERMS

CCC	core component container
FFTF	Fast Flux Test Facility
ISA	interim storage area
LWR	light water reactor
NRC	U.S. Nuclear Regulatory Commission
NSNF	National Spent Nuclear Fuel
OCRWM	Office of Civilian Radioactive Waste Management
QARD	quality assurance requirements and description
RL	U.S. Department of Energy, Richland Operations Office
SNF	spent nuclear fuel

D14.0 QUALITY ASSURANCE

D14.1 INTRODUCTION

An introduction to the quality assurance program that includes the objectives and scope that apply to all Spent Nuclear Fuel (SNF) Project quality assurance activities is provided in Chapter 14.0 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1.

D14.2 REQUIREMENTS

The requirements that form the basis of the quality assurance program are identified in Section 14.2 of HNF-3553, Volume 1. Additional requirements for the 200 Area Interim Storage Area (ISA) are identified in the following paragraphs. These requirements ensure that the quality assurance requirements for federal repository acceptance of SNF and U.S. Nuclear Regulatory Commission (NRC) equivalency requirements are satisfied. Tables D4-1 and D4-2 present listings of safety-class and safety-significant structures, systems, and components required for the 200 Area ISA.

The U.S. Department of Energy, Richland Operations Office (RL), has directed (Sellers 1995) that DOE/RW-0333P, *Quality Assurance Requirements and Description for the Civilian Radioactive Waste Management Program (QARD)*, published by the Office of Civilian Radioactive Waste Management (OCRWM), be applied as the principal quality assurance document to the SNF Project OCRWM program. RL has directed application of DOE/RW-0333P to the following SNF-related activities as they relate to repository storage:

- Characterization or data collection for input and use
- Conditioning into final form
- Handling, packaging, and transportation.

Items, activities, and documentation determined to be important to safety are presented in Table 3-1 and Section 5.0 of HNF-SD-SNF-RPT-007, *Application of the Office of Civilian Radioactive Waste Management (OCRWM) Quality Assurance Requirements to the Hanford Spent Nuclear Fuel Project*. This document identifies structures, systems, components, and activities that require application of DOE/RW-0333P requirements to ensure compliance with the RL direction. Assumptions described in Section 2.2 of HNF-SD-SNP-RPT-007, applicable to the ISA and QARD compliance, include the following:

- 324 Building Light Water Reactor (LWR) SNF will be repackaged from the LWR canister prior to repository acceptance.

- Neutron Radiography Facility TRIGA[1] casks will not be accepted by OCRWM.

- Fast Flux Test Facility (FFTF) SNF will be repackaged from core component containers (CCC) prior to repository acceptance.

[1] TRIGA is a trademark of General Dynamics Corporation.

- Recommendations regarding compliance with the QARD are independent of the need to comply with Title 10, *Code of Federal Regulations*, Part 830, "Nuclear Safety Management," Subpart A, "Quality Assurance Requirements;" Title 10, *Code of Federal Regulations, Part 71*, "Packaging and Transportation of Radioactive Material," Subpart H; Title 10, *Code of Federal Regulations*, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Subpart G; or other documents specified by the SNF Project.

- Where an activity, such as data collection, and presentation have been completed outside an approved OCRWM QARD program, the data identification, review adequacy, and usage will be validated and qualified in accordance with OCRWM Supplement III requirements.

Quantities of site-wide SNF inventories, the current storage location, and information related to each type of fuel is provided in Table 5-1 and Figure 5-1 of HNF-SD-SNF-RPT-007. Application of the QARD requirements to these fuels must address process and storage plans for each fuel type.

In addition, specific quality requirements include: (1) periodic surveillance inspections of the SNF during storage; (2) records of all inspections after acceptance of the SNF by the SNF Project; (3) record(s) of fuel damage or other nonconformances and corrective actions after acceptance of the SNF by the SNF Project; and (4) developing and maintaining OCRWM data records for the SNF inventory, as identified in formal direction from RL. Data received by the SNF Project will be considered unqualified and tracked as such until the data passes an appropriate qualification process per the QARD.

As stated in Section 5.2.2 of HNF-SD-SNF-RPT-007, the SNF Project will develop a disposition plan for Hanford Site SNF that will identify the compliance strategy for each SNF inventory. Additionally, the SNF Project will have lead responsibility for the review of guidance on final disposition requirements from the National Spent Nuclear Fuel (NSNF) Program and will provide input to the NSNF Program on the acceptability of the draft guidance.

DOE has established a regulatory policy (Grumbly 1995) that new SNF Project facilities involved in processing K Basins SNF will achieve nuclear safety equivalency with NRC-licensed facilities. An evaluation, documented in WHC-SD-SNF-DB-002, *Spent Nuclear Fuel Project Path Forward, Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities*, identified requirements to establish nuclear safety equivalency that are to be met in addition to existing and applicable DOE requirements. These requirements, except those related to the design basis earthquake, are contained in HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements*. HNF-SD-SNF-DB-004, *Spent Nuclear Fuel Project Seismic Design Criteria NRC Equivalency Evaluation Report*, contains the design basis earthquake requirements. The SNF Project has self-imposed some of these requirements to the ISA.

NRC nuclear safety equivalency requirements identified in HNF-SD-SNF-DB-003 that will be applied to the ISA include the following:

- Review and approval by RL of changes to HNF-MP-599, *Project Hanford Management System Quality Assurance Program Description*, that could be interpreted as decreasing the quality assurance program's existing commitments for the ISA (HNF-SD-SNF-DB-003, Item 16).

- Ensure that the appropriate quality requirements in existing Project Hanford Management Contract procedures and instructions, as identified in WHC-SD-SNF-DB-002, remain in effect (HNF-SD-SNF-DB-003, Item 18).

Design requirements for natural phenomena hazards, other than seismic design requirements, are identified in HNF-2524, *200 East Area Interim Storage Area Preliminary Safety Evaluation*.

The documents cited in this chapter identify the requirements to achieve nuclear safety equivalency with NRC-licensed facilities and to meet the requirements of DOE/RW-0333P. The quality assurance program plan for SNF Project facilities provides for implementation of these requirements. A graded approach will be used for items and activities important to safety (i.e., safety-class, safety-significant, and certain general-service items and/or activities), in accordance with HNF-MP-599 and NUREG/CR-6407, *Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety.*

D14.3 QUALITY ASSURANCE PROGRAM ORGANIZATION

A summary of the SNF Project quality assurance program, including summaries of safety management policies and philosophies used as a basis for the program, is provided in Section 14.3 of HNF-3553, Volume 1.

The SNF Project organizational structure, responsibilities, authorities, and interfaces that apply to the ISA are addressed in Chapter 17.0 of HNF-3553, Volume 1.

D14.4 QUALITY IMPROVEMENT

Descriptions of SNF Project management programs and processes used to correct adverse conditions affecting quality at all SNF Project facilities are provided in Section 14.4 of HNF-3553, Volume 1.

D14.5 DOCUMENTS AND RECORDS

A description of the SNF Project document control and records management program associated with quality assurance is provided in Section 14.5 of HNF-3553, Volume 1.

D14.6 QUALITY ASSURANCE PERFORMANCE

An overview of the SNF Project process to ensure that the performed work meets requirements is provided in Section 14.6 and its subsections in HNF-3553, Volume 1. The subsections address work processes, design activities, the procurement process, program tests and inspections, management assessments, and independent assessments.

D14.7 REFERENCES

10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*.

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

10 CFR 830, "Nuclear Safety Management," Subpart A, "Quality Assurance Requirements," *Code of Federal Regulations*, as amended.

DOE/RW-0333P, 2000, *Quality Assurance Requirements and Description for the Civilian Radioactive Waste Management Program*, Rev. 10, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, Washington, D.C.

Grumbly, T. P., 1995, *Concurrence with the K Basin Spent Nuclear Fuel Project Policy on Nuclear Safety Requirements* (Memorandum EM-36-3.1.6.7 to Manager, U.S. Department of Energy, Richland Operations Office, July 20), U.S. Department of Energy, Washington, D.C.

HNF-2524, 1998, *200 East Area Interim Storage Area Preliminary Safety Evaluation*, Rev. 0, DE&S Hanford, Incorporated, Richland, Washington.

HNF-3553, 2000, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Vol. 1, Rev. 0-B, Fluor Hanford, Incorporated, Richland, Washington.

HNF-MP-599, 2001, *Project Hanford Management System Quality Assurance Program Description*, Rev. 7, Fluor Hanford, Incorporated, Richland, Washington.

HNF-SD-SNF-DB-003, 1998, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements*, Rev. 4-A, Fluor Daniel Northwest, Incorporated, Richland, Washington.

HNF-SD-SNF-DB-004, 1997, *Spent Nuclear Fuel Project Seismic Design Criteria NRC Equivalency Evaluation Report*, Rev. 2A, Fluor Daniel Hanford, Incorporated, Richland, Washington.

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CHAPTER D15.0

EMERGENCY PREPAREDNESS PROGRAM

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CONTENTS

LIST OF TABLES

LIST OF TERMS

BED	building emergency director
CSB	Canister Storage Building
EPZ	emergency planning zone
ISA	interim storage area
RL	U.S. Department of Energy, Richland Operations Office
SCBA	self-contained breathing apparatus
SNF	spent nuclear fuel

D15.0 EMERGENCY PREPAREDNESS PROGRAM

D15.1 INTRODUCTION

A description of the philosophy, objectives, and organization of the Spent Nuclear Fuel (SNF) Project emergency preparedness program for response to emergencies at the SNF Project facilities is provided in Chapter 15.0 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1, and Chapter A15.0, Annex A. This Annex D chapter presents emergency management information specific to the 200 Area Interim Storage Area (ISA).

D15.2 REQUIREMENTS

The requirements that form the basis for the SNF Project emergency preparedness program are identified in Section 15.2 of HNF-3553, Volume 1.

D15.3 SCOPE OF EMERGENCY PREPAREDNESS

Potential ISA emergencies could span the spectrum of identified emergencies for SNF Project facilities, from worker injuries to general emergencies with potential public impact. The spectrum of emergencies that the ISA emergency preparedness program is designed to encompass is described in Section 15.3 of HNF-3553, Volume 1, and in Chapter D3.0. Chapter 4.0 of HNF-2524, *200 East Area Interim Storage Area Preliminary Safety Evaluation*, presents a discussion of the principal accidents that could occur at the ISA.

D15.4 EMERGENCY PREPAREDNESS PLANNING

SNF Project emergency preparedness planning includes identification of emergency organizations, assessment actions, notification processes, emergency facilities and equipment, protective actions, access control, training, drills, exercises, and recovery actions. A summary of the emergency response organization that is activated for ISA emergencies is provided in Section 15.4 of HNF-3553, Volume 1, and Section A15.4, Annex A. ISA emergencies and responses will be covered in the Canister Storage Building (CSB) building emergency plan and emergency response procedures.

The provisions in Section 15.4.1.2 of HNF-3553, Volume 1, are fulfilled by the building emergency director (BED) at the CSB for the ISA until an Incident Command Post is established. The shift manager is the BED for both onsite hazardous and nonhazardous facilities, as stated in Section 15.4.1.2 of HNF-3553, Volume 1.

D15.4.1 Emergency Response Organization

Section 15.4.1 of HNF-3553, Volume 1, presents information related to the organizational structure, building emergency organization, Incident Command Post, Emergency Operations Center, and support resources to meet emergency event requirements. This information also applies to the ISA.

The CSB Emergency Response Organization is responsible for establishing an Incident Command Post to respond to events occurring at the ISA facility (Section 15.4.1.2 of HNF-3553, Volume 1). Event response information is conveyed to and from the U.S. Department of Energy, Richland Operations Office (RL) Emergency Operations Center. DOE-RL maintains responsibility for communication with offsite agencies, as indicated in Figure 15-2 in HNF-3553, Volume 1. Each hazardous SNF Project facility has an emergency staff of individuals who assist in the protection of personnel, property, and the environment. Initial direction and control of an emergency response at the ISA is the responsibility of the BED prior to establishment of the Incident Command Post. Key Emergency Response Organization positions and responsibilities are discussed in the subsections that follow.

D15.4.1.1 Building Emergency Director

The BED, as described in Section 15.4.1.2 of HNF-3553, Volume 1, and Section A15.4.1.2, Annex A, is the emergency coordinator for hazardous/facility-related events and has the authority to commit all SNF Project resources (equipment and personnel) in response to any emergency and to request supporting resources. Other responsibilities include: (1) implementing a building emergency plan; (2) assuring that the ISA Emergency Response Organization is fully staffed and trained; (3) initially assessing, categorizing, and classifying events; (4) notifying the Patrol Operations Center and applicable contractor and DOE management through the Occurrence Notification Center; (5) implementing protective actions; (6) establishing an initial ISA Incident Command Post; (7) controlling the event scene; (8) initiating mitigating activities; and (9) initiating recovery actions when directed. At the ISA, the BED is a certified CSB operations shift manager.

A listing of the primary and alternate BEDs by title, work location, and work telephone numbers is contained within the ISA building emergency plan. The BED is on the CSB or ISA premises during hazardous operations and is available through an "on-call" list 24 hours a day at all other times. Operations maintains a listing of on-call BED names, with work and home telephone numbers, at the Occurrence Notification Center.

D15.4.1.2 Incident Command Post Staff

The Incident Command Post staff is a group of SNF Project emergency response personnel assigned to an Incident Command Post established for an event. The Incident Commander, as supported by the BED, Hanford Fire Department, and Hanford Patrol Shift Commander, directs all emergency response efforts at the event scene.

Emergency response efforts for the Hanford Site are conducted by the Incident Commander, BED, and Hanford Patrol Shift Commander. The BED becomes a member of the Incident Command Post and functions under the direction of the Incident Commander. In this role, the BED continues to manage and direct ISA operations.

D15.4.1.3 Event Scene Staff

The event scene staff is composed of a Hanford Fire Department Operations Section Chief (assigned by the Incident Commander), trained support staff (including Health Physics and Industrial Hygiene staff), and (as required) Hanford Fire Department medical responders and Hanford Patrol. In addition, accountability aides are responsible for facilitating the implementation of protective actions (evacuation or take cover) and for facilitating the accountability of personnel after the protective actions have been implemented. Staging area managers are responsible for coordinating/conducting activities at the staging area. Personnel accountability aides assist the staging managers by ensuring that personnel and visitors are properly evacuated from designated staging areas to a safe location. The event scene staff, as directed by the Incident Command Post, supports actions requested by the Incident Commander and the BED.

D15.4.2 Assessment Actions

Provisions of Section 15.4.2 of HNF-3553, Volume 1, cover hazards survey, hazards assessment, emergency action levels, consequence assessment, and monitoring activities. These provisions apply to the ISA.

An ISA emergency planning hazards assessment will be developed for the ISA for hazards that have the potential to generate an "Alert" or higher emergency. The hazards assessment will be prepared from the hazard and safety analyses that are developed and included in Chapter D3.0. The hazards assessment will also be derived from other pertinent facility documentation (e.g., safety assessment documents, interim safety basis documents, and special nuclear material accountability documents). The hazards assessment provides the technical basis for the emergency management program. The scope and extent of planning and preparedness directly corresponds to the type and scope of hazards present and the potential consequences of events.

The hazards assessment identifies and characterizes the hazards relevant to potential ISA operational emergencies. This includes determination of the following:

- A broad range of initiating events
- Accident mechanisms
- Equipment or system failures
- Event indications
- Contributing events
- Source terms
- Material release characteristics
- Topography

- Environmental transport and diffusion
- Exposure considerations
- Chemical hazards.

The hazards assessment characterizes the potential consequences to workers, the public, and the environment for each postulated accident and determines the emergency planning zone (EPZ) for each facility. The assessment also determines the emergency class, protective actions, and observable indications and criteria (emergency action levels) corresponding to the range of identified accidents.

A spectrum of potential accidents ranging from minor to beyond design basis are postulated and realistically analyzed. While not every conceivable situation will be analyzed, the hazards assessments provide the framework for response to virtually any declared emergency.

The methodology, assumptions, models, and evaluation techniques used in the hazards assessments are documented in Sections 15.4.2.3 and 15.4.2.4 of HNF-3553, Volume 1. Results from the ISA hazards assessment are used to develop the ISA building emergency plan elements contained in the CSB building emergency plan. Hazards assessments for the ISA are reviewed annually and updated, as necessary, in accordance with Section 15.4.2.2 of HNF-3553, Volume 1.

D15.4.3 Notification

Notifications, in the event of an emergency event at the ISA, will be made in accordance with the provisions of Section 15.4.3 of HNF-3553, Volume 1, in order to mitigate consequences and to protect the health and safety of workers, the public, and the environment.

D15.4.4 200 Area Interim Storage Area Emergency Facilities and Equipment

The building emergency plan for the ISA is part of the CSB building emergency plan. A description of the facilities available for coordinating ISA emergency response activities is specified in the building emergency plan.

Emergency equipment consisting of materials and tools that may be required to measure, control, or mitigate the consequences of an emergency at the ISA is provided in Table D15-1. Detection ranges and types of instruments for radiological and nonradiological hazardous materials are adequate for ISA emergency conditions, as determined in Section D15.4.2. The emergency planning organization ensures that sufficient emergency equipment is available. The location of this emergency equipment is stated in the CSB building emergency plan.

Table D15-1. Interim Storage Area Emergency Equipment.

Type of equipment	Equipment capabilities
Fixed and portable equipment	
Fire protection systems – Fire detection and alarm system and dry pipe automatic sprinkler suppression system	Assists in notifying personnel, summoning the Hanford Fire Department, and in fire suppression
Fire system pressure alarms and/or water flow alarm	Assists with notifying personnel of emergency conditions
Evacuation and take cover siren	Assists with notifying personnel of emergency conditions and, by the type of siren, expected actions
Respiratory protection (SCBA)[a]	Protects personnel from hazardous chemicals
Portable emergency equipment	
Fire extinguishers (Types A, B, and C)	Assists in fire suppression
Hazardous materials spill control kits (unmounted)	Assists with hazardous (chemical) materials stabilization and cleanup following a spill or release
Command post equipment: emergency procedures, checklists (maps and photographs of facilities optional)	Provides area and site-specific emergency information
Operational event scene equipment: radiological response vehicle, emergency procedures, duty cards, checklists, maps, photographs of facilities	Assists in controlling and mitigating the event
Protective clothing and equipment	
Anti-C clothing and personal protective equipment	Provides contamination control (anti-C clothing for radiological and acid gear for any corrosive chemicals)
Miscellaneous respiratory equipment	Provides respiratory protection for radionuclides; this type of respirator equipment is not considered to be emergency equipment

[a] SCBA respirators for emergency use will be thoroughly inspected at least once a month and after each use. Records of inspection dates and findings will be maintained.

SCBA = self-contained breathing apparatus.

D15.4.5 Protective Actions

Protective actions are those actions taken to preclude or reduce the exposure of individuals or the environment impacted by hazards or unsafe conditions during an emergency event at the ISA. These protective actions are presented in Section 15.4.5 of HNF-3553, Volume 1, and are applicable to the ISA. Protective actions for the ISA will reflect use of the emergency response planning guidelines identified in Section 15.4.2.3 of HNF-3553, Volume 1. The planning guidelines published in the *Emergency Response Planning Guidelines* (AIHA 1988) will be used during an ISA emergency response to determine protective actions for unique exposures to chemical releases (see Table 15-4 of HNF-3553, Volume 1). The protective action guides are also used during an emergency response to determine protective actions for unique exposures to radiological releases (see Table 15-1 of HNF-3553, Volume 1). In DOE/RL-94-02, *Hanford Emergency Management Plan*, RL directs the use of the published protective action guides adopted by the states of Washington and Oregon from EPA-100, *Manual of Protective Action Guides and Protective Actions for Nuclear Accidents*.

The Hanford Site emergency management program uses the EPZ concept to focus emergency planning activities. EPZs are designated areas where protective actions may be required. The size of a zone is determined primarily by the expected dispersion distance of a particular concentration of a substance. The two exposure pathways for both radiological and nonradiological hazardous materials are the plume exposure pathway and the ingestion exposure pathway. A description of the exposure pathways is provided in Section 15.4.5 of HNF-3553, Volume 1. Figure D1-1 indicates the location of the CSB. Figure 2-1 in HNF-2524 shows the location of the ISA relative to the CSB.

The plume exposure pathway EPZ is the probable area of exposure to a passing cloud (or plume) of the substance, potentially resulting in direct contact with the substance through the exterior of the body or through inhalation. The plume exposure pathway EPZ includes the area where emergency planning is conducted (1) to ensure that prompt and effective actions are taken in the event of an emergency, (2) to protect onsite personnel, and (3) to ensure public health and safety. The plume exposure pathway for the CSB (10 mi) is shown in Figure 15-6 in HNF-3553, Volume 1.

The ingestion exposure pathway EPZ is the probable area of exposure to contaminated foodstuffs or water potentially resulting in deposition of the material in various internal organs following ingestion (eating or drinking). The ingestion exposure pathway EPZ for radiological and nonradiological incidents at all Hanford Site facilities corresponds to the 80-km (50-mi) EPZ for the Energy Northwest (formerly known as Washington Public Power Supply System) Columbia Generating Station. The gray area in Figure 15-6 in HNF-3553, Volume 1, represents the ingestion EPZ for the Hanford Site.

The protective actions required to minimize the exposure of workers and the public are summarized in Section 15.4.5 of HNF-3553, Volume 1. Examples of protective actions as a function of accident category and consequences are illustrated in Table 15-5 in HNF-3553, Volume 1.

D15.4.6 Training and Exercises

The CSB emergency organization will be formed, trained, and tested for potential ISA emergency events in accordance with the provisions of Section 15.4.6 of HNF-3553, Volume 1. Drills and exercises will be developed in accordance with Section 15.4.6 of HNF-3553, Volume 1, with sufficient scope and detail to emphasize the facility-specific emergency events and response actions applicable to the ISA.

D15.4.7 Reentry and Recovery

The provisions applicable to a ISA emergency event termination, facility entry, transition from an emergency organization to a recovery organization, and the recovery process are provided in Section 15.4.7 of HNF-3553, Volume 1.

D15.5 DOCUMENT CONTROL

The ISA building emergency plan, implementing procedures, reports of drills and exercises, and emergency event documentation will be controlled and updated in accordance with the provisions of Section 15.5 of HNF-3553, Volume 1.

D15.6 REFERENCES

AIHA, 1988, *Emergency Response Planning Guidelines*, American Industrial Hygienist Association, Akron, Ohio.

DOE/RL-94-02, 2001, *Hanford Emergency Management Plan*, Rev. 2, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

EPA-100, 1992, *Manual of Protective Action Guides and Protective Actions for Nuclear Accidents*, U.S. Environmental Protection Agency, Washington, D.C.

HNF-2524, 1998, *200 East Area Interim Storage Area Preliminary Safety Evaluation*, Rev. 0, DE&S Hanford, Incorporated, Richland, Washington.

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CHAPTER D16.0

PROVISIONS FOR DECONTAMINATION AND DECOMMISSIONING

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CONTENTS

LIST OF TERMS

D&D	decontamination and decommissioning
ISA	interim storage area
SNF	spent nuclear fuel

D16.0 PROVISIONS FOR DECONTAMINATION AND DECOMMISSIONING

D16.1 INTRODUCTION

The provisions that apply to future decontamination and decommissioning (D&D) of Spent Nuclear Fuel (SNF) Project facilities are addressed in Chapter 16.0 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1. Provisions specific to the 200 Area Interim Storage Area (ISA) are addressed in this Chapter D16.0.

The 200 Area ISA has been given the designation of a Hazard Category 2 facility because of the inventory contained in the cask storage systems. The facility will not experience significant, if any, contamination during interim storage of the SNF. Therefore, decontamination efforts at the end of facility life will only be required in the event that an unanticipated release occurs during the life of the facility. Decommissioning efforts will also be minimal based on the simplicity of the facility design (e.g., concrete slabs, graveled storage and access areas, a fence, and lighting fixtures). The end state of the 200 Area ISA is not known at this time, but several options, ranging from facility reuse to complete dismantlement and removal, are available.

D16.2 REQUIREMENTS

The requirements that form the basis for the D&D program are found in Section 16.2 of HNF-3553, Volume 1. Specific requirements applicable to this chapter include the following:

- DOE Order 5820.2A, *Radioactive Waste Management*, Chapter V, "Decommissioning of Radioactively Contaminated Facilities."
- DOE Order 6430.1A, *General Design Criteria*, Division 13, "Special Facilities," Section 1300, "General Requirements," paragraph 1300-11, "D&D," and Section 1320, "Irradiated Fissile Material Storage Facilities," paragraph 1320-7, "D&D."
- Title 10, *Code of Federal Regulations*, Part 835, "Occupational Radiation Protection" (10 CFR 835), Section 835.1002(d).

D16.3 DESCRIPTION OF CONCEPTUAL PLANS

General D&D considerations applicable to all SNF Project facilities are provided in Section 16.3 of HNF-3553, Volume 1.

D16.3.1 Design Features

Each 200 Area ISA cask storage configuration provides confinement of radiological materials during transport of the SNF to the 200 Area ISA and during storage at the ISA. The cask storage configuration design features that are important to D&D will be controlled through the design change control process to ensure that changes to the facility will provide equal or greater consideration to D&D. Further description of the confinement design is provided in Chapter D2.0.

D16.3.1.1 Safety Features

Chapters D3.0, D4.0, D5.0, and D6.0 identify the selected safety features for preventing or mitigating the postulated accidents. These implemented engineered barriers and administrative programs will prevent or greatly reduce the consequences of any postulated upset event or accident; thus, the amount and location of any radiological material releases will be diminished for D&D cleanup activities. No design basis accident will result in offsite dose consequences (prevented) or onsite dose consequences exceeding the evaluation guidelines (mitigated).

D16.3.2 Operational Considerations

The potential for personnel or equipment contamination is minimized by the design of the dry cask storage systems, and by administrative controls, radiological practices, and work guidelines defined in operating procedures and work permits. Because baseline operations assume no spread of contamination from the cask storage systems, no special facilities for the support of decontamination activities have been provided. Although the risk of contamination is minimal, operating procedures address requirements for radiological control surveys in the facility. For example, surveys will be performed annually to verify that each cask system remains free of contamination.

D16.3.3 Decommissioning

The 200 Area ISA components may be removed at a future time in compliance with applicable regulations. The process used to develop the 200 Area ISA D&D plan is provided in Section 16.3.3 of HNF-3553, Volume 1. Conceptual plans for D&D will include an updated facility hazard analysis for the D&D activities, which will be used to prepare the plan to administer the expected D&D strategy. Conceptual plans will also include a preliminary deactivation plan that contains at least the following information:

- Structures, systems, and components in their final configuration
- Review and determination of the status of structures, systems, and components based on the mission and life-cycle phase

- Configuration management for missing or inaccurate design baseline documentation, voiding and downgrading of design documents, and turnover of design baseline documents to the environmental restoration contractor.

Decommissioning plans for the 200 Area ISA facility will be developed and reviewed against the existing environmental impact statement. The environmental impact statement will be updated to include D&D activities in accordance with the *National Environmental Policy Act (NEPA) of 1969* process, if deemed appropriate.

The construction of the 200 Area ISA as an above-grade facility will simplify decontamination and dismantling. The ISA storage building will be decontaminated as necessary and dismantled using conventional techniques. The ISA concrete pads will be dismantled and removed from the site.

D16.4 REFERENCES

10 CFR 835, "Occupational Radiation Protection," *Code of Federal Regulations*, as amended.

DOE Order 5820.2A, 1988, *Radioactive Waste Management*, U.S. Department of Energy, Washington, D.C.

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

HNF-3553, 2000, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Vol. 1, Rev. 0-B, Fluor Hanford, Incorporated, Richland, Washington.

National Environmental Policy Act (NEPA) of 1969, 42 USC, 4321, et seq.

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CHAPTER D17.0

MANAGEMENT, ORGANIZATION, AND INSTITUTIONAL SAFETY PROVISIONS

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CONTENTS

LIST OF TERMS

ISA	interim storage area
SNF	spent nuclear fuel

D17.0 MANAGEMENT, ORGANIZATION, AND INSTITUTIONAL SAFETY PROVISIONS

D17.1 INTRODUCTION

A description of the organizational structure, responsibilities, and interfaces that support safe design, construction, and operational activities of the 200 Area Interim Storage Area (ISA), as a subproject of the Spent Nuclear Fuel (SNF) Project, are addressed in Chapter 17.0 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Volume 1.

D17.2 REQUIREMENTS

The requirements that form the basis for management, organizational, and safety provisions are identified in Section 17.2 of HNF-3553, Volume 1.

D17.3 ORGANIZATIONAL STRUCTURE, RESPONSIBILITIES, AND INTERFACES

The overall organizational structure, responsibilities, and interfaces for ISA operations are identified in Section 17.3 of HNF-3553, Volume 1.

D17.4 SAFETY MANAGEMENT POLICIES AND PROGRAMS

The safety management policies and programs applicable to ISA are identified in Section 17.4 of HNF-3553, Volume 1.

D17.5 REFERENCES

HNF-3553, 2000, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Vol. 1, Rev. 0-B, Fluor Hanford, Incorporated, Richland, Washington.

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