**This document was too large to scan
as a single document. It has
been divided into smaller sections.**

Section 1 of 3

Document Information			
Document #	SD-SNF-SARR-005		REV 2
Title	MULTI-CANISTER OVERPACK (MCO) TOPICAL REPORT		
Date	05/11/2000		
Originator	B.D. LORENZ,K.E. SMITH,S.F. KESSLER,D.M. CHENAULT,L.H. GOLDMANN,C.R. HOOVER,D.R. DUNCAN,B. KNUTSON,G.D. BANKEN,Y.J. LIU	ORIGINATOR CO.	XWEST GROUP, INC.,BABCOCK SERVICES,FFS, FH,NHC,PARS ONS INFR. & TECHNOLOGY Q-METRICS
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ENGINEERING CHANGE NOTICE

Page 1 of _____

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ECN

2. ECN Category (mark one) Supplemental ☒ Direct Revision ☐ Change ECN ☐ Temporary ☐ Standby ☐ Supersedure ☐ Cancel/Void ☐		3. Originator's Name, Organization, MSIN, and Telephone No. B.D. Lorenz / SNF-Nuclear Safety / R3-26 / 372-3529		4. USQ Required? ☒ Yes ☐ No		5. Date	
		6. Project Title/No./Work Order No. Spent Nuclear Fuel Project		7. Bldg./Sys./Fac. No.		8. Approval Designator E, S, Q, D	
		9. Document Numbers Changed by this ECN (includes sheet no. and rev.) HNF-SD-SNF-SARR-005, Rev 1		10. Related ECN No(s). N/A		11. Related PO No. N/A	
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13a. Description of Change Revision 2 of hte MCO Topical Report was prepared for the following reasons: To address MCO Topical Report SER open items To address OCRWM requirements To update references To remove unnecessary detail from Chapter 1 figures To achieve consistency with revised SARP Analysis To make minor editorial changes USQ# K-00-0336 USQ# MCO-00-027 USQ# CVD-00-0077							
14a. Justification (mark one) Criteria Change ☒ Design Improvement ☐ Environmental ☐ Facility Deactivation ☐ As-Found ☐ Facilitate Const. ☐ Const. Error/Omission ☐ Design Error/Omission ☐		14b. Justification Details The MCO Topical Report required modification in several areas to address the SER open items. While making those revisions, it was decided to incorporate the OCRWM discussion (Chapter 13) to achieve consistency with the latest thermal and gas generation analysis and to update references. Figure simplification was also performed to lighten the USQ load.					
15. Distribution (include name, MSIN, and no. of copies) Control Distribution						RELEASE STAMP MAY 11 2000 DATE: STA: 4 HANFORD RELEASE ID: S	

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Page 2 of 2

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

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16. Design Verification Required

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☒ No

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Savings ☐ \$

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Improvement ☐ N/A

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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	☐		☐
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

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	Signature	Date
Design Agent		
PE		
QA		
Safety		
Design		
Environ.		
Other		

DEPARTMENT OF ENERGY

Signature or a Control Number that tracks the Approval Signature

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S

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Revision 2

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Total Pgs 694
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Multi-Canister Overpack Topical Report

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Assistant Secretary for Environmental Management

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

Fluor Hanford
P.O. Box 1000
Richland, Washington

5/10/00
Release Approval Date

MAY 11 2000		
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MULTI-CANISTER OVERPACK TOPICAL REPORT

**HNF-SD-SNF-SARR-005
Revision 2**

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March 2000

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LIST OF TERMS

ALARA	as low as reasonably achievable
ARF	airborne release fraction
ARR	airborne release rate
ASHRAE	American Society of Heating, Ventilation, and Refrigeration Engineers
ASME	American Society of Mechanical Engineers
CSB	Canister Storage Building
CVDF	Cold Vacuum Drying Facility
DBA	design basis accident
DOE	U.S. Department of Energy
DOE-RL	U.S. Department of Energy, Richland Operations Office
EPRI	Electric Power Research Institute
FH	Fluor Hanford, Incorporated
FSAR	Final Safety Analysis Report
HEPA	high-efficiency particulate air
ICRU	International Commission on Radiation Units and Measurements
MAR	material at risk
MCNP	Monte Carlo N-particle
MCO	multi-canister overpack
MHM	multi-canister overpack handling machine
MTU	metric ton of uranium
NIST	National Institute of Standards and Technology
NPH	natural phenomena hazard
NRC	U.S. Nuclear Regulatory Commission
PCB	polychlorinated biphenyl
pcm	percent milli-k
PMF	probable maximum flood
PMP	probable maximum precipitation
RF	respirable fraction
SAR	safety analysis report
SARP	safety analysis report for packaging
SNF	spent nuclear fuel
SSC	structure, system, and component
SPR	single pass reactor
t	metric ton
TSR	technical safety requirement

1.0 GENERAL DESCRIPTION

1.1 INTRODUCTION

In February 1995, the U.S. Department of Energy (DOE) approved the Spent Nuclear Fuel (SNF) Project's "Path Forward" recommendation for resolution of the safety and environmental concerns associated with the deteriorating SNF stored in the Hanford Site's K Basins (Hansen 1995). The recommendation included an aggressive series of projects to design, construct, and operate systems and facilities to permit the safe retrieval, packaging, transport, conditioning, and interim storage of the K Basins' SNF. The facilities are the Cold Vacuum Drying Facility (CVDF) in the 100 K Area of the Hanford Site and the Canister Storage Building (CSB) in the 200 East Area. The K Basins' SNF is to be cleaned, repackaged in multi-canister overpacks (MCOs), removed from the K Basins, and transported to the CVDF for initial drying. The MCOs would then be moved to the CSB and weld sealed (Loscoe 1996) for interim storage (about 40 years).

One of the major tasks associated with the initial Path Forward activities is the development and maintenance of the safety documentation. In addition to meeting the construction needs for new structures, the safety documentation for each must be generated. A common thread that was identified among the structures was the MCO. Each structure exists for the specific purpose of treating or storing the MCO and its contents. Normally, an extensive amount of MCO-related documentation would be generated for each of the facility safety analysis reports. However, the expedited schedule for removing spent fuel from the K Basins requires that the documentation effort be minimized and repetitious activities be eliminated. Therefore, this topical report has been prepared to address those aspects of the MCO that will be common to the facilities. The MCO will be included in each facility's safety documentation by reference to this topical report. By capturing the design of the MCO and its safety evaluation in a single document, repetition, inconsistency, and duplication of effort is minimized. As changes are made to the MCO, the MCO Topical Report can be updated to retain the validity of references to the report. By submitting the topical report for early review and approval, it will be possible for the facility safety analysis reports to reference the Topical Report for much of the MCO design information.

Use of topical reports is not new to the nuclear industry. The U.S. Nuclear Regulatory Commission (NRC) has historically used a Topical Report Program, primarily for the review and approval of hardware, methodology, or analyses that are proposed for use throughout the industry. The topical report was developed to provide a detailed discussion of designs and analyses with sufficient detail to enable the regulatory agency to make a determination that such designs and analyses were acceptable and could be applied generically to different facilities provided that facility-specific evaluation be performed and approved by design and approval authorities at that facility. The topical report approach is a methodology that significantly reduces repetitious review and provides a significant cost savings. The approach is simply a one-time NRC review and approval, with plant-specific application by utilities after evaluation of the topical. The topical is incorporated into the utilities documentation by reference thereafter. Such an application can represent significant savings in the preparation of the SNF Project safety

documentation. In accordance with the above, this MCO Topical Report has been prepared to provide a description of MCO design aspects and applicable analyses and evaluations.

There is limited substantive guidance relative to the design or fabrication of the primary storage container 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), or in NRC Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)* (NRC 1989b). However, NRC Regulatory Guide 3.61, *Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask* (NRC 1989a), contains a significant amount of guidance that is appropriate for the MCO. Regulatory Guide 3.61 (NRC 1989a) therefore has been used for guidance in developing the format and content of this MCO Topical Report. Since direct applicability to the MCO did not exist in all areas, several modifications to the guidance were made to address the differences between the MCO and a storage cask. Regulatory Guide 3.61 (NRC 1989a) addresses a dry storage cask design that is essentially a complete self-contained system. The MCO, however, does not function independently, as it is continually dependent upon other systems or facility features. Despite the differences, this regulatory guide does provide the most direct NRC guidance applicable to the MCO. This topical report was prepared utilizing this direct guidance, and incorporating pertinent modifications. The result is a source of MCO-related information that can be readily referenced in the safety documentation for SNF processing and storage facilities.

1.2 GENERAL DESCRIPTION OF THE MULTI-CANISTER OVERPACK

The SNF MCO is a single-use container that consists of a cylindrical shell, five or six fuel baskets, a shield plug, and design features necessary for maintaining the structural integrity of the MCO while providing criticality control and fuel drying capability. The MCO design and assembly features are depicted in Figure 1-1. HNF-0426, *Performance Specification for Spent Nuclear Fuel Multi-Canister Overpack* (Goldmann 2000b) implements the functional requirements identified and documented in WHC-SD-SNF-FRD-016, *Spent Nuclear Fuel Multi-Canister Overpack Technical Functions and Requirements* (Goldmann 1996a), defines the MCO, and thus establishes the minimum essential requirements that the MCO design must meet.

The MCO provides confinement of SNF and maintains the SNF in a critically safe configuration. NRC requirements (NRC 1996) for a 10 CFR 72 licensed facility would allow for application of either Subsection NB or NC, 3.IV.1.b(1)(b) of the *Boiler and Pressure Vessel Code* (ASME 1998). The MCO's structural design has been developed in accordance with the *Boiler and Pressure Vessel Code* (ASME 1998), Section III, "Rules for Construction of Nuclear Power Plant Components," Subsection NB. Deviations from these rules in the structural design or analyses are documented and justified.

1.2.1 Shell

The MCO shell is a stainless steel (304L) cylindrical vessel with access at the top end that is closed with a stainless steel shield plug. The shell is fabricated from 24-in.- diameter, schedule 80S pipe with a wall thickness of 0.5 in. (Figure 1-2), and an overall length of 160 in., without the cover cap, when assembled with the shield plug assembly (Figure 1-1). The MCO cavity is approximately 23 in. in diameter and 140 in. long. The MCO has a machined bottom plate that has a constant thickness of 2.0 in. except in the center region, where it is 1.13 in. thick (Figure 1-3). The MCO's bottom plate and basket support plates allow free water to move to the long process tube pickup point for removal. The vessel holds the fuel, fuel fragments, fuel baskets, and incidental equipment. Incidental equipment includes criticality prevention features such as the center post tube nesting feature on the shield plug and bottom plate, shield plug process ports, internal filters, and sealing features. The design of the MCO vessel meets service level A requirements for normal operating loads and service level D requirements for accident conditions under the Boiler and Pressure Vessel Code, Section III, Subsection NB (ASME 1998).

1.2.2 Fuel Baskets

All baskets are annular open-top containers with a nominal outer dimension of 22.6 in. All baskets will support the fuel at 1.0 g for temperatures up to 132 °C (270 °F). All basket designs incorporate a center post tube for axial support during lifting and for protection of a long process tube.

Each basket is loaded, in the upright position, by the fuel retrieval system equipment in the K Basin pool; stored, upright, in a loading queue; and loaded into the MCO by the MCO loading system. The loaded baskets can be easily and safely handled in the basin water, reliably loaded and nested into the cask-MCO assembly in the K Basin load-out pits, and engaged with the shield plug guard plate and axial stabilizer.

The baskets stack inside the MCO with the baskets' centerlines coincident with the MCO's centerline. While stacked inside the MCO, the baskets provide for insertion of a long process tube down the MCO centerline for water draining and gas transport, as needed. The baskets can drain freely and will not capture or retain excessive water during bulk water removal at the CVDF.

Basket design accounts for differential thermal expansion when subjected to elevated temperatures inside the MCO. The baskets support heat transfer into and out of the fuel while in the gaseous and vacuum environments inside the MCO. The primary heat transfer modes are radiation and conduction during the static (storage/monitoring) state.

The baskets are designed to provide pathways for the gas flows needed to properly dry intact fuel and scrap fuel during cold vacuum drying. The baskets are compatible with the fuel and MCO materials during the expected temperatures, pressures, and atmospheres inside the MCO during handling, drying, shipping, and storage.

The MCO fuel baskets are categorized into two major types: intact fuel element baskets and scrap fuel (fragment) baskets (Figures 1-4 through 1-7). The Mark IA fuel and scrap baskets are designed in accordance with the Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME 1998) under the component safety group as guided by NUREG/CR 3854, *Fabrication Criteria for Shipping Containers* (NRC 1984). Mark IA fuel has a higher ^{235}U enrichment than Mark IV fuel. Structural integrity is required of the Mark IA basket for criticality control whereas structural integrity for criticality control is not required for the Mark IV fuel basket. Therefore, the more stringent Section III, Subsection NG requirements (ASME 1998) are applied to the construction of the Mark IA fuel and scrap baskets.

As specified, the Mark IV fuel and scrap baskets are not required to be designed to the Boiler and Pressure Vessel Code (ASME 1998) because the Mark IV fuel has a lower ^{235}U enrichment than Mark IA fuel. The Mark IV baskets are designed and fabricated using stress allowables from the Boiler and Pressure Vessel Code, Section III, Subsection NG. Safety factors of three for material yield and five for material failure are applied, as this is the common industry standard for rigging and lifting operations around personnel. These requirements were based upon good engineering practices and judgement. Although not imposed on the project, ANSI N14.6-1986, *For Radioactive Materials — Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More* (ANSI 1986), is the source of the three and five requirement. Only applicable portions of ANSI N14.6-1986 have been used. Analyses indicate that the Mark IV fuel cannot achieve criticality in an MCO under normal operating conditions or accident scenarios. It follows that the basket's structural integrity is not required for Mark IV criticality control. Design allowables per the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME 1998) and safety factors described below were utilized. However, during accident conditions the baskets designed for Mark IA fuel and for Mark IA fuel scrap shall maintain the criticality control features in accordance with Chapter 6.0. For handling of both loaded and unloaded Mark IA and Mark IV baskets, the designs meet safety factors of three on material yield and five on material failure.

The Mark IV and Mark IA fuel and scrap basket structural portions are made of 304L stainless steel. Scrap basket materials also include copper for added thermal conductivity. The baskets maintain their structural integrity during expected internal MCO environmental conditions, normal MCO handling situations, and after accidents (Mark IA baskets only). The baskets are sufficiently strong to preserve their integrity during bulk water removal and vacuum drying under normal MCO handling conditions, for various internal MCO environments, and after MCO design basis accidents (Mark IA baskets only).

The baskets do not have gas-producing materials that significantly increase the pressure in the MCO during storage. The baskets also do not have materials that appreciably accelerate corrosion or significantly alter the properties of the MCO confinement boundary.

The screws used in the outer posts are safety class for Mark IA baskets. The spring pin used for the centerpost to baseplate thread is general service because it is only a backup to prevent loosening of the centerpost. Similarly, the flow restrictors, even if eliminated, would have only a minor effect on scrap temperatures.

1.2.3 Shield Plug

The MCO shield plug is made of 304L stainless steel and is designed to mate with the open end of the MCO shell (Figure 1-8). The shield plug protects workers from photons and neutrons emanating from the inside of the MCO. This shielding design maintains an average dose rate across the top of the shield plug of 2 mrem/h on contact (2 in.) for the average MCO fuel inventory. The average 2 mrem/h dose rate includes radiation streaming between the MCO shield plug and MCO shell and streaming around penetrations; radiation streaming between the MCO and cask is not included. For the worst-case MCO, the average dose rate across the top of the shield plug on contact (2 in.) does not exceed 8 mrem/h. Actual doses are much lower, as indicated in Chapter 5.0.

The shield plug is closed using a mechanical closure assembly. The mechanical closure assembly holds the shield plug in place using a threaded locking ring that is put into the MCO collar after the shield plug is inserted. Once assembled, the 18 set screws in the locking ring are tightened down into the shield plug's back side to compress the silver-plated seal between the MCO shell and the shield plug. The MCO shield plug assembly also mates with the end effector on the top SNF fuel basket. The MCO has a minimum height of approximately 1.0 in. of free space between the top of the SNF materials and the bottom of the guard plate on the shield plug at room temperature. This gap is not safety related but simply a design feature.

The shield plug provides access to the interior of the MCO via four penetrations (Figure 1-9). The penetrations accommodate two process ports integral to the shield plug (ports 2 and 3), a safety-class rupture disk port (port 4), which also serves as a backup port for vacuum drying at the CVDF, and a spare, plugged port (port 1). The two process ports have valve mechanisms to accommodate connections to external equipment (Figure 1-10). These process ports connect internally to two process paths. One process path (port 3) connects to a long, 1-in. schedule XXS pipe that extends down the MCO's axis to the bottom, and the second process path (port 2) (1 in. diameter) extends to internal filters mounted on the shield plug's underside. A third path (port 4) (1 in. diameter) extends from the rupture disk to the space between the bottom of the shield plug and the top of the guard plate. The long, 1-in. schedule XXS process pipe has a 2-mm debris screen around the end (Figure 1-9). The connections leading to the two process tubes are designed to be easily differentiated by a worker looking at either the top or bottom of the shield plug. The design of the penetrations, ports, and valve mechanism implements the following operational and performance criteria:

- Provisions for pressurizing the MCO interior with an inert gas
- Provisions for purging gases from the MCO interior
- Provisions for leak rate testing, where applicable
- Provisions for making or breaking all connections while continuing to maintain SNF confinement with minimal spread of contamination
- Connections designed to facilitate their decontamination

- Ports and connections accessible to the operator from the top face of the MCO
- Penetrations and connections that do not appreciably reduce or impair MCO shielding
- Provisions for removal or reinstallation of sealing mechanisms as required to cover process connections (these sealing mechanisms, including fasteners, cannot extend above the top of the shield plug [Figure 1-11]). The Helicoflex¹ seals can be replaced at the K Basins if necessary.
- Provisions for controlled bleed down of internal MCO pressure into the process piping
- Penetrations and connections designed to facilitate remote operation via long-handled tools and via a manipulator if needed.

The shield plug features an integrally machined axisymmetric lifting ring with a 12-ton lifting capacity when gripped with six equally spaced grippers. The ring facilitates handling of the MCO package with the MCO handling machine (MHM). Design of the MHM lifting rim area of the MCO locking and lifting ring must include a safety factor of three on material yield and five on material ultimate strength. As previously noted, safety factors of three for material yield and five for material failure are applied, as this is the common industry standard for rigging and lifting operations around personnel. These requirements were based upon good engineering practices and judgement. Although not imposed on the project, ANSI N14.6-1986 (ANSI 1986), is the source of the three and five requirement. Only applicable portions of ANSI N14.6-1986 have been used.

1.2.4 Additional Features

The MCO has internal filters to support the vacuum drying outflows from the MCO (Figure 1-12). These filters are installed between the shield plug bottom and the guard plate. The internal high-efficiency particulate air (HEPA) filters are present primarily for process reasons and general facility cleanliness. The filters also provide the benefit of reducing potential contamination even though no credit is taken for such a function. These internal filters are HEPA filters and adequate flow capacity is achieved through the filter bank to support operating needs. These filters do not meet the provisions of DOE Order 6430.1A, *General Design Criteria*, because their efficiency cannot be tested after the shield plug is installed. The filters are not relied upon to perform a safety function and are present for process purposes only. No credit has been taken for the presence or function of these filters in the accident analyses associated with any of the facilities. Port number 4 serves as an unfiltered backup process port at the CVDF.

¹Helicoflex is a trademark of Helicoflex Corporation.

To summarize the key features, the MCO is a stainless steel container that is mechanically sealed to a stainless steel shield plug. On top of the shield plug are four ports. The ports access four penetrations, two of which are used for processing (ports 2 and 3). The long tube (port 3) allows water pickup and gas transport as needed. Port 4 doubles as the location for the safety-class pressure relief device (rupture disk) and as an unfiltered backup process port.

The remaining two ports (ports 1 and 2) are connected to a bank of four internal HEPA process filters. Port 2 is a filtered process port used during vacuum drying operations and port 1 is a plugged port with potential to be used during MCO monitoring activities or future gas sampling activities. The ports are all capable of being sealed using bolted cover plates. Chapter 8.0 provides a detailed discussion of the MCO port configurations. Eight screened holes (2-mm metal mesh) in the guard plate functionally expands the blowdown surface area available to the rupture disk by approximately a factor of eight, making plugging of the disk substantially less likely than without the screen.

The ASME Boiler and Pressure Vessel Code states that the operating area (surface area) of the pressure relief device may not be reduced. The screened holes in effect increase the surface area to receive raw flow.

1.2.5 Human Factors Design Considerations

The MCO is designed for interface with operators at various stages of assembly and operation. Human factors considerations have been incorporated into the MCO design and are manifested in the overall integrated handling, assembly, and use of the MCO components. All movable and replaceable components are located on the outside of the MCO and are accessible from the outside. These external components are designed to support the fuel loading, sealing, transporting, processing, and sampling operations planned for the MCO.

Materials are "tracked" during fabrication of all MCO pieces. When MCO components are assembled, the appropriate numbering is applied to the discreet unit. Serial numbers are placed on the shell assembly (i.e., the collar, shell wall, and bottom cap), the shield plug assembly, the locking and lifting ring, the cover cap, and the individual baskets. The labeling plan has been agreed to in advance by operations personnel and has been found to be suitable and appropriate for the function of the unit.

The fuel baskets are designed so operators can see the individual pockets in the baskets during fuel retrieval operations. The pockets are located in a fuel rack plate, and whole fuel elements are placed into these pockets. The design allows an operator to easily recognize which pockets are filled and which are open.

The shield plug lifting fixture is designed to minimize operator strain during the handling and placement of the shield plug. The assembly equipment used to insert and tighten the long axial process tube includes a weight-balancing cup to minimize operator strain and wrench flats on the tube to positively assist in manual tightening of the long axial process tube. The design of the

locking and lifting ring also considered the handling machine interface and allowed features for a counter-balanced gripper design. The handling machine for the ring allows for threading the locking and lifting ring into the collar with minimal force supplied by the operator.

Much of the interface with the assembled MCO occurs on the operating face of the shield plug. Features such as cross-drilled appliances allow for the bleed down of pressure without removal of the appliance. All appliances include 1/3125-in. standard hexagonal interfaces for commonality. The interface with the shield plug also allows for the use of remote tools and operators, if needed. Cone-head bolts are used on the cover plates to allow easy placement of the socket tools whether the operator is working directly or remotely. The ports have also been numbered to avoid any confusion on the part of operators.

Finally, the placement and welding of the cover cap is operator friendly. The cover cap has an integral centering-backing ring built into the cap design, so the placement and centering of the cover cap on the MCO collar at the CSB work stations is easy. The MCO cover cap design also allows for easy placement, setup, and alignment of the welding machine for the cap-to-collar attachment weld as the machine is indexed from the cover cap.

1.3 MULTI-CANISTER OVERPACK CHARACTERISTICS

1.3.1 Design and Fabrication

The MCO components will be designed, fabricated, inspected, and examined in accordance with the Boiler and Pressure Vessel Code (ASME 1998), Section III, Subsection NB, except for the Mark IA spent fuel basket assemblies, which must meet Subsection NG. All MCO welds performed by the offsite fabricator will be completed and examined in accordance with the Boiler and Pressure Vessel Code (ASME 1998), Section III, Subsection NB, as well as the final cover cap weld. Exceptions to NB and other subsections, if any, are discussed in the final design report (Goldmann 2000a).

The following structural materials are used to fabricate the MCO:

- ASME SA-182, type 304N stainless steel for the locking and lifting ring
- ASME SA-182, dual certified F304/F304L stainless steel for the MCO collar
- ASME SA-312, type 304L stainless steel for the MCO shell and shield plug
- ASME SA-182, type 304L stainless steel for the bottom baseplate assembly
- ASME SA-193, grade B8S or B8SA stainless steel set screws
- Silver-plated stainless steel for Helicoflex closure seal.

The mechanical properties of the MCO's structural materials are presented in Section 3.2. The listed properties are Boiler and Pressure Vessel Code (ASME 1998) minimums and in some instances higher property usage may be justified. The expected weights and centers of gravity of the MCO, its contents, and ancillary equipment after drying also are presented in Section 3.2. The operational features of the MCO will be limited to the lifting attachments, closure design, pressure relief connections, access connections, and the internal bank of four HEPA filters located on the underside of the shield plug.

1.3.2 Confinement Boundary

The MCO is to be the interim storage container for the spent fuel and is relied upon as the primary confinement boundary (see Section 1.2). The MCO confinement boundary consists of the following items (Figure 1-1):

- Shield plug
- MCO collar
- MCO shell
- MCO bottom baseplate
- Weld connecting shell to bottom and to collar
- Shell longitudinal weld
- Shield plug to shell mechanical closure and Helicoflex seal (before cover cap welding)
- Four port cover flanges and C-seals (before cover cap welding)
- Port cover closure bolts (before cover cap welding)
- MCO cover cap (when the MCO is seal welded as shown on Figure 1-1)
- Rupture disk in shield plug port number 4 (not always active); port is covered with flange before transporting to CSB).

The MCO shield plug assembly will be mechanically sealed to maintain a low leakage rate. Figure 1-1 provides the details of the mechanical closure assembly. A containment weld will be made to attach the MCO shell to the MCO bottom baseplate assembly and collar. Welds will be made in conformance with the Boiler and Pressure Vessel Code (ASME 1998), Section III, Subsection NB. All welds are examined in accordance with the Boiler and Pressure Vessel Code (ASME 1998), Section III, Subsection NB-5300. Details of the MCO locking and lifting ring as

well as the MCO cover cap are provided in Figure 1-13. Each port may be covered by a blind flange port cover that can maintain a leaktight seal

The MCO cover caps are installed in the CSB work stations. The MCO is sampled, if necessary, and the port appliances and covers are checked to ensure they are properly configured for MCO interim storage at the CSB. All MCO mechanical seals are leakage tested at the CVDF before shipping. Leakage rates are specified in Section 9.1.1.2. Optional testing is used only when leakage is suggested as a consequence of some damage occurring to the MCO. The weld preparation areas of the collar are cleaned and inspected to make sure the area is clean and free of deleterious conditions or contamination that may affect the weld's properties. A cover cap is positioned on the MCO collar.

The weld machine is brought in, and the inert weld backer gas tube is introduced underneath the cover cap through the penetration. The purge of air begins from the newly created chamber with the introduction of the inert backer gas. Welding begins with a root pass. The completed root pass is manually dye penetrant examined and then more passes of weld are laid down. Another dye penetrant examination may be performed at the halfway point for the weld. The final weld passes are made, and the weld machine is removed, with the inert purge continuing and the inerting of the chamber preserved. The weld is leakage rate tested and dye penetrant examined for the last time in accordance with ASME Code Case 595. The penetration plug is tightened and leakage rate tested. The penetration is eventually permanently closed with the 2.25-in.-diameter cover plate disk that is welded into the cover cap penetration hole.

When the cover cap is on the MCO, the cover cap penetration allows helium to be put under the cover cap so the attachment weld may be leakage rate tested using a test collar on the outside and the flow pumped to a helium mass spectrometer (this is an essential function of the cover cap penetration)

The cover cap penetration also has the following potential uses:

1. Putting the welding backer gas under the cover cap during welding attachment to the collar; this reduces oxidation of the molten weld and surrounding heat-affected metals
2. With helium below the shield plug, testing the leakage rate in the MCO Helicoflex main seal by vacuum pumping on the cover cap cavity and sending the flow to a helium mass spectrometer
3. Pressure load testing the cover cap after welding installation of the cover cap onto the collar; this would be done by pressurization of the cavity created by the cover cap installation
4. Allowing access through the cover cap penetration to operate the number 2 port appliance in the shield plug for gas sampling, pressure adjustment, and/or exchange inside the MCO.

1.4 MULTI-CANISTER OVERPACK CONTENTS

The fuel to be stored in the MCOs is located in the K Basins storage pools. The 105 K East fuel storage basin contains 51,073 fuel assemblies (some are not intact), elements, or pieces. The 105 K West fuel storage basin contains 53,964 fuel assemblies, elements, or pieces. The total mass of the fuel elements at the 105 K East fuel storage basin is approximately 1,233 metric ton (t); the total mass of the fuel elements in the 105 K West fuel storage basin is approximately 1,038 t. This total includes approximately 1,143,600 kg of uranium and 2,155 kg of plutonium in the 105 K East fuel storage basin and approximately 951,900 kg of uranium and 1,875 kg of plutonium in the 105 K West fuel storage basin.

The cladding on a significant amount of the fuel, due to normal handling, was damaged during discharge from the N Reactor or during subsequent fuel handling. Video imaging indicates that the outer elements as well as the inner elements have breached or missing sections of cladding. As a result, the uranium in some elements was exposed to water and has oxidized during storage. This oxidation causes the fuel elements to swell, leading to further cladding damage, which exposes fresh uranium to the basin water and further oxidation. The loss of cladding integrity and the oxidation of the uranium allows soluble and gaseous fission products to dissolve into the basin and canister water.

The fuel assemblies and scrap (pieces greater than 0.25 in. in all dimensions) will be contained in five or six baskets (depending on fuel type), placed one on top of another, in the vertical MCO. The baskets will have an optimum carrying capacity in terms of weight as well as criticality considerations. More information regarding basket structural design and analyses is presented in Section 3.0 of this report.

1.5 IDENTIFICATION OF AGENTS AND CONTRACTORS

For purposes of this report, the Owner is the DOE and the Buyer is Fluor Hanford, Inc. (FH).

Agents and contractors responsible for the design and fabrication of the MCO and the baskets are as follow:

- FH

Responsibility: FH is responsible for the overall management and execution of the MCO subproject. This includes development of the subproject functions and requirements, performance specification, safety documentation, design review and approval, quality assurance, procurement, budget, schedule, and coordination with interfacing subprojects.

- Parsons Infrastructure and Technology Group, Inc.

Responsibility: Parsons Infrastructure and Technology Group, Inc., was the original design agent for the MCO subproject and was tasked with development of the MCO design in accordance with the performance specification. Parsons was responsible for the professional quality, technical accuracy, quality assurance, and cost-effectiveness of the design media to be used for fabrication of the MCOs.

Responsibility for design administration, including changes to the design, now rests with FH.

- Joseph Oat Corporation

Responsibility: Joseph Oat Corporation is responsible for materials, fabrication, welding, examination, shop testing, quality assurance, documentation, ASME code stamping, packaging, and shipping for the empty MCOs in accordance with the requirements of the fabrication specification. This includes development of a manufacturing plan, schedule, fabrication drawings, material certificates, fabrication and welding procedures, examination reports, and test reports to verify that the materials and completed work conform to the fabrication specifications.

- FH, with support from DynCorp Tri-Cities Services, Inc.

Responsibility: FH, with support from DynCorp Tri-Cities Services, Inc., is responsible for materials, fabrication, welding, examination, shop testing, quality assurance, documentation, packaging, and shipping for the MCO baskets in accordance with the requirements of the fabrication specification. This includes development of a manufacturing plan, schedule, fabrication drawings, material certificates, fabrication and welding procedures, examination reports, and test reports to verify that the materials and completed work conform to the fabrication specifications.

1.6 U.S. NUCLEAR REGULATORY COMMISSION EQUIVALENCY

In *K Basins Spent Nuclear Fuel Project - Regulatory Policy*, dated August 4, 1995 (Sellers 1995), the DOE established the requirement for new SNF Project facilities to achieve "nuclear safety equivalency" to comparable NRC-licensed facilities. Given this policy, the SNF Project performed a review and evaluation of Title 10 of the *Code of Federal Regulations* and relevant NRC guidance against the existing DOE requirements for new SNF Project facilities. The purpose of the review and evaluation was to identify any additional actions, beyond the existing DOE requirements, that were necessary to demonstrate nuclear safety equivalence. The result of this review was the identification of several NRC requirements that were necessary to achieve nuclear safety equivalency. These requirements could be met by the MCO or by the facilities in which the MCO resides. These requirements are documented in

HNF-SD-SNF-DB-005, *Spent Nuclear Fuel Project Multi-Canister Overpack, Additional NRC Requirements* (Garvin 1998). Appendix 1A contains an assessment matrix listing the NRC requirements for the MCO and corresponding compliance statements.

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Figure 1-1. Multi-Canister Overpack.
(sheet 1 of 7)

Figure 1-1. Multi-Canister Overpack.
(sheet 2 of 7)

Figure 1-1. Multi-Canister Overpack.
(sheet 4 of 7)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-1. Multi-Canister Overpack.
(sheet 5 of 7)

GUARD PLATE, RING &
PROCESS TUBE ASSY

Figure 1-1. Multi-Canister Overpack.
(sheet 6 of 7)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-1. Multi-Canister Overpack.
(sheet 7 of 7)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-2. Multi-Canister Overpack Shell.

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-3. Multi-Canister Overpack Shell Bottom Plate Subassembly.

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-4. Mark IV Spent Fuel Storage Basket. (sheet 1 of 4)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-4. Mark IV Spent Fuel Storage Basket. (sheet 2 of 4)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-4. Mark IV Spent Fuel Storage Basket. (sheet 3 of 4)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-4. Mark IV Spent Fuel Storage Basket. (sheet 4 of 4)

Figure 1-5. Mark 1A Spent Fuel Storage Basket. (sheet 1 of 5)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-5. Mark IA Spent Fuel Storage Basket. (sheet 2 of 5)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-5. Mark 1A Spent Fuel Storage Basket. (sheet 3 of 5)

Figure 1-5. Mark IA Spent Fuel Storage Basket. (sheet 4 of 5)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-5. Mark IA Spent Fuel Storage Basket. (sheet 5 of 5)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-6. Mark IV Spent Fuel Scrap Basket. (sheet 1 of 5)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-6. Mark IV Spent Fuel Scrap Basket. (sheet 2 of 5)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-6. Mark IV Spent Fuel Scrap Basket. (sheet 3 of 5)

Figure 1-6. Mark IV Spent Fuel Scrap Basket. (sheet 4 of 5)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-6. Mark IV Spent Fuel Scrap Basket. (sheet 5 of 5)

ALL DIMENSIONS ARE NOMINAL.
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-7. Mark IA Spent Fuel Scrap Basket. (sheet 1 of 6)

Figure 1-7. Mark IA Spent Fuel Scrap Basket. (sheet 2 of 6)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-7. Mark IA Spent Fuel Scrap Basket. (sheet 3 of 6)

Figure 1-7. Mark IA Spent Fuel Scrap Basket. (sheet 4 of 6)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-7. Mark IA Spent Fuel Scrap Basket. (sheet 5 of 6)

1A BASE PLATE, SCRAP

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-7. Mark IA Spent Fuel Scrap Basket. (sheet 6 of 6)

1A SHROUD CELL SUBASSEMBLY

ALL DIMENSIONS ARE NOMINAL.
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-8. Multi-Canister Overpack Mechanical Closure Shield Plug.
(sheet 1 of 3)

SHIELD PLUG SUBASSEMBLY

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-8. Multi-Canister Overpack Mechanical Closure Shield Plug.
(sheet 2 of 3)

Figure 1-8. Multi-Canister Overpack Mechanical Closure Shield Plug.
(sheet 3 of 3)

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-9. Schematic Diagram of Multi-Canister Overpack Passages.

Port number	Function
1	Filtered backup port
2	Filtered process exit port
3	Long axial process tube
4	Safety-class rupture disk and backup process port

Figure 1-10. Multi-Canister Overpack Process Valves.
(sheet 1 of 2)

Figure 1-10. Multi-Canister Overpack Process Valves.
(sheet 2 of 2)

PROCESS VALVE — PORT 2
HEPA FILTER

PROCESS VALVE — PORT 3
LONG PROCESS TUBE

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-11. Multi-Canister Overpack Process Port Cover Plates.

4-HOLE COVER PLATE ASSEMBLY SIMILAR

5-HOLE COVER PLATE ASSEMBLY SHOWN

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-12. Multi-Canister Overpack Integrated Filter Assembly.

FILTER SUBASSEMBLY

ALL DIMENSIONS ARE NOMINAL
FOR EXACT DIMENSIONS, REVERT
TO CONTROLLED MCO AND
BASKET DESIGN DRAWINGS.

Figure 1-13. Multi-Canister Overpack Cover Cap and Locking and Lifting Ring.
(sheet 1 of 2)

Figure 1-13. Multi-Canister Overpack Cover Cap and Locking and Lifting Ring.
(sheet 2 of 2)

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APPENDIX 1A

**ASSESSMENT OF MULTI-CANISTER OVERPACK COMPLIANCE WITH
SPENT NUCLEAR FUEL PROJECT U.S. NUCLEAR REGULATORY
COMMISSION EQUIVALENCY CRITERIA**

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APPENDIX 1A

**ASSESSMENT OF MULTI-CANISTER OVERPACK COMPLIANCE WITH SPENT NUCLEAR FUEL PROJECT
U.S. NUCLEAR REGULATORY COMMISSION EQUIVALENCY CRITERIA**

Item No.	Additional NRC requirements as identified in HNF-SD-SNF-DB-005, Rev. 3	Related DOE/Project Hanford references	Assessment of compliance for MCO
1	<p>Ensure the containment/confinement integrity of the MCO and its subcriticality geometry are preserved when the design basis earthquakes for the CSB and CVDF, as outlined in WHC-SD-SNF-DB-004, are imposed on these facilities.</p> <p>(References: 10 CFR 72.24, "Contents of application: Technical information," 10 CFR 72.90, "General considerations," 10 CFR 72.92, "Design basis external natural events," 10 CFR 72.102, "Geological and seismological characteristics," 10 CFR 72.122, "Overall requirements," and 10 CFR 72.212, "Conditions of general license issued;" Regulatory Guide 1.60, <i>Design Response Spectra for Seismic Design of Nuclear Power Plants</i>, Rev. 1; Regulatory Guide 1.61, <i>Damping Values for Seismic Design of Nuclear Power Plants</i>, Rev. 0; Regulatory Guide 3.48, <i>Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)</i>, Rev. 1; and SECY-93-087, <i>Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs</i>)</p>	<p>DOE 5480.28 and companion standards DOE 6430.1A HNF-PRO-097</p>	<p>This requirement is satisfied by the CSB and CVDF designs. Details regarding the natural phenomena loads and applicable DOE, NRC, and national consensus standards applied to the facility designs are documented in the facility FSARs (HNF-3553, Annex A and Annex B).</p> <p>CVDF design for natural phenomena, consistent with the NRC equivalency criteria, is documented in the CVDF Annex of the SNF Project FSAR, Chapters 1.0, 2.0, and 4.0, and in supporting references (e.g., WHC-SD-SNF-DB-010). The CVDF facility annex, <i>Chapter 4.0 summarizes natural phenomena loads and applicable DOE, NRC, and national consensus standards applied to the CVDF for seismic events, straight wind, tornado, volcanic ash, flooding, ash, flooding, lightning, and snow. The CVDF requirements for natural phenomena are implemented in the design by the design requirements document (HNF-SD-SNF-DRD-002) and the controlling lower tier design and procurement documentation.</i></p> <p>CSB design for natural phenomena, consistent with the NRC equivalency criteria, is documented in the CSB Annex of the SNF Project FSAR, Chapters 1.0, 2.0, and 4.0, and in supporting references (e.g., WHC-SD-SNF-DB-009). The CSB facility annex, <i>Chapter 4.0 summarizes natural phenomena loads and applicable DOE, NRC, and national consensus standards applied to CSB for seismic events, straight wind, tornado, volcanic ash, flooding, lightning, and snow. The CSB requirements for natural phenomena are implemented in the design by the CSB Performance Specification (WHC-S-0425) and the controlling lower tier design and procurement documentation.</i></p>

Item No.	Additional NRC requirements as identified in HNF-SD-SNF-DB-005, Rev. 3	Related DOE/Project Hanford references	Assessment of compliance for MCO
2	<p>Ensure the containment/confinement integrity of the MCO and its subcritical geometry are preserved when a design basis tornado (including translational velocity, rotational velocity, and pressure differential) and tornado missile are imposed on the designs of the CSB and CVDF, taking into consideration Regulatory Guide 1.76, <i>Design Basis Tornado for Nuclear Power Plants</i>, SECY-93-087, <i>Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs</i>, and NUREG/CR-4461, <i>Tornado Climatology of the Contiguous United States</i>.</p> <p>(Reference: 10 CFR 72.122, "Overall requirements")</p>	DOE 5480.28 and companion standards DOE 6430.1A HNF-PRO-097	This requirement is satisfied by the CSB and CVDF designs. Details regarding the natural phenomena loads and applicable DOE, NRC, and national consensus standards applied to the facility designs are documented in the facility annexes of the SNF Project FSAR.
3	<p>Ensure the design of the MCO supports ready retrieval.</p> <p>(Reference: 10 CFR 72.122, "Overall requirements")</p>	DOE 6430.1A	Sections 4.11 and 4.13 of the MCO Performance Specification (HNF-S-0426) address requirements for lifting and handling the MCO to permit safe retrieval of the MCO from the CSB storage tubes. The MCO closure design accommodates ready access to the SNF by removal of the cover cap and disassembly of the mechanical closure.
4	<p>Use Regulatory Guide 1.26 to assist in assigning the appropriate ASME <i>Boiler and Pressure Vessel Code</i>, Section III, code classes to the MCO shell, parts, and subassemblies, as applicable. (NUREG/CR-3854, <i>Fabrication Criteria for Shipping Containers</i>, has direct application to the MCO and may be used in lieu of Regulatory Guide 1.26.)</p> <p>(Reference: Regulatory Guide 1.26, <i>Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants</i>, Rev. 3)</p>	DOE 6430.1A	After review of both Regulatory Guide 1.26 and NUREG/CR-3854, the MCO design authority choose to apply NUREG/CR-3854 for assigning an ASME code class because NUREG/CR-3854 more directly applied to the MCO. The MCO Performance Specification (HNF-S-0426), Section 4.9.1, invokes application of Section III, Subsection NB, of the ASME <i>Boiler and Pressure Vessel Code</i> as guided by NUREG/CR-3854.

Item No.	Additional NRC requirements as identified in HNF-SD-SNF-DB-005, Rev. 3	Related DOE/Project Hanford references	Assessment of compliance for MCO
5	<p>Review the NRC positions in Regulatory Guide 1.84 and Regulatory Guide 1.85 on <i>ASME Boiler and Pressure Vessel Code</i>, Section III, code cases before using such code cases for safety-class applications for the MCO. The NRC positions on applicable code cases should be used in the design. Where no NRC position is stated in regards to the acceptance of a code case, that code case may be used as approved by the Code Committee.</p> <p>(References: Regulatory Guide 1.84, <i>Design and Fabrication Code Case Acceptability ASME Section III, Division I</i>, Rev. 30; and Regulatory Guide 1.85, <i>Materials Code Case Acceptability ASME Section III, Division I</i>, Rev. 30)</p>	DOE 6430.1A	ASME code case N-284 is applied to the MCO shell as outlined in Section 13.2 of Appendix 5 of the MCO Design Report (HNF-SD-SNF-DR-003). This code case is acceptable for use according to Regulatory Guide 1.84. Code case N-595, pending approval by ASME, will be referenced in the next revision to the MCO Design Report.

Item No.	Additional NRC requirements as identified in HNF-SD-SNF-DB-005, Rev. 3	Related DOE/Project Hanford references	Assessment of compliance for MCO
6	<p>Incorporate a requirement into safety-class procurement specifications for the MCO that requires suppliers to report defects and noncompliance in items or services. The requirement should be similar to the following.</p> <p>Safety-class equipment and/or services furnished under this order are subject to reporting of defects. If equipment and/or services contain defects that could cause a substantial safety hazard, then immediate reporting to the Buyer is required unless the Seller has actual knowledge that the Buyer has been adequately informed of such defect.</p> <p>The Seller shall evaluate identified or suspected defects. If the Seller's evaluation determines a defect does exist that could cause a substantial safety hazard, then the Seller shall notify the Buyer as soon as practicable and, in all cases, within 5 working days following completion of the evaluation.</p> <p>If the Seller determines that it does not have the capability to perform the evaluation, the Seller may request the Buyer to cause an evaluation to be performed. Seller's request shall be effected within 5 working days of this determination. If the Seller elects to have the Buyer perform the evaluation, then all necessary and pertinent information and correspondence shall be sent to the Buyer. Results of evaluations by the Buyer will be transmitted to the Seller.</p> <p>(Reference: 10 CFR 21, "Reporting of Defects and Noncompliance")</p>	HNF-PRO-298	<p>The MCO fabrication contract will contain the following words (currently used for the contract on the first three MCOs (contract MGW-SLX-A31483, clause B22Xf)):</p> <p>"Safety class equipment and/or services furnished under this order are subject to reporting of defects. If equipment and/or services contain defects which could cause a substantial safety hazard, then immediate reporting to the Buyer is required, unless the Seller has actual knowledge that the Buyer has been adequately informed of such defect.</p> <p>"The Seller shall evaluate identified or suspected defects. If the Seller's evaluation determines a defect does exist which could cause a substantial safety hazard, then the Seller shall notify the Buyer as soon as practicable, and in all cases, within five (5) working days following completion of the evaluation.</p> <p>"If the Seller determines that it does not have the capability to perform the evaluation, the Seller may request the Buyer to cause an evaluation to be performed. Seller's request shall be effected within five (5) working days of this determination. If the Seller elects to have the Buyer perform the evaluation, then all necessary and pertinent information and correspondence shall be sent to the Buyer. Results of evaluations by the Buyer will be transmitted to the Seller.</p> <p>"The requirements for reporting of defects as described above are in effect beginning with contract award and remain in force throughout the projected forty (40) year life of the safety class equipment furnished under this contract."</p>

Item No.	Additional NRC requirements as identified in HNF-SD-SNF-DB-005, Rev. 3	Related DOE/Project Hanford references	Assessment of compliance for MCO
7	<p>Before implementation, DOE-RL will review and approve any changes to WHC-SP-1131, <i>Quality Assurance (Q4) Program and Implementation Plan</i>, for the SNF Project that could be interpreted as decreasing the Quality Assurance Program's existing commitments for the SNF Project (HNF-SP-1228, <i>Quality Assurance Program Implementation Plan for Nuclear Facilities</i>, is in the process of being revised for subsequent approval by DOE-RL. Nuclear facility lists and descriptions from WHC-SP-1131 are being relocated to ES&H nuclear safety documents. WHC-SP-1131 remains in effect until DOE-RL approval is secured for HNF-SP-1228 and facility lists and descriptions are relocated to ES&H nuclear safety documents.)</p> <p>(References: 10 CFR 50.54(a), "Conditions of licenses [Quality Assurance Provisions]")</p>	10 CFR 830.120	The MCO Topical Report (HNF-SD-SNF-SARR-005), Chapter 13.0, documents the quality assurance requirements for the MCO subproject, including reference to the SNF Quality Assurance Program Plan (SNFP-1) and Quality Assurance Program Implementation Plan for Nuclear Facilities (HNF-SP-1228), which implement the requirements of 10 CFR 830.120 for the SNF Project. These plans require DOE approval prior to any changes, similar to the NRC rule 10 CFR 50.54(a).
8	<p>Ensure for MCO fabrication that the areas of vendor and subcontractor quality assurance records and control of safety-class purchased material, equipment, and services receive emphasis during SNF Project audits, surveillances, and assessments.</p> <p>(Reference: IEN 95-29)</p>	Project Hanford procurement clauses as appropriate	Control of vendor and subcontractor quality assurance records and control of safety-class purchased SSCs are verified via scheduled and unscheduled surveillances performed to verify vendor and subcontractor compliance with established contractual quality assurance requirements as well as their SNF Project-approved quality assurance programs. Additional SNF Project quality assurance surveillances are scheduled to verify the fabricators' compliance with contract quality assurance program requirements and their SNF Project-approved quality assurance programs as fabrication of MCOs proceeds toward completion and delivery. Surveillances performed to date during fabrication of the first 30 baskets and three training and testing MCOs are available for review.
9	<p>Implement the WHC Occurrence Reporting System for the design and fabrication of the MCO.</p> <p>(Reference: 10 CFR 50.55(e), "Conditions of construction permits")</p>	DOE 5000.3B HNF-PRO-060	Section 9.2.6 of the MCO Fabrication Specification (HNF-S-0453) requires reporting of all nonconformances in materials, approved procedures, design drawings, and purchase documents. Application of the Project Hanford Occurrence Reporting System is an in-house action that the SNF Project would take should design or fabrication flaws be reported or discovered. Reporting of SNF Project "occurrences" during the design and construction phase is implemented by administrative procedure AP 2-15.

Item No.	Additional NRC requirements as identified in HNF-SD-SNF-DB-005, Rev. 3	Related DOE/Project Hanford references	Assessment of compliance for MCO
10	<p>Use a criticality safety value of 0.95 for k_{eff} for MCO design, functions, and related activities.</p> <p>(References: NUREG-0800, 1981, <i>Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants</i>, SRP 9.1.2, "Spent Fuel Storage," Rev. 3; and NUREG-0612, <i>Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36</i>)</p>	<p>DOE 5480.24 HNF-PRO-334 HNF-PRO-537 HNF-PRO-540 HNF-PRO-544 HNF-PRO-545</p>	<p>The criticality safety evaluation report for spent fuel processing and storage facilities (HNF-SD-SNF-CSER-005) documents analyses of MCO criticality safety that demonstrate the k_{eff} will not exceed 0.95. This information is also documented in Chapter 6.0 of the MCO Topical Report (HNF-SD-SNF-SARR-005). The criticality control measures necessary to ensure a k_{eff} of less than 0.95 relative to the MCO design are documented in Section 4.19.3 of the MCO Performance Specification (HNF-S-0426).</p>

Item No.	Additional NRC requirements as identified in HNF-SD-SNF-DB-005, Rev. 3	Related DOE/Project Hanford references	Assessment of compliance for MCO
11	<p>Identify SSCs "important to safety" in accordance with 10 CFR 72.3. Once SSCs are identified as having a function meeting the definition of important-to-safety, impose the requirements for SSCs important to safety specified in 10 CFR 72.</p> <p>A graded approach is applied to an SSC important to safety by using the guidance provided in NUREG/CR-6407, <i>Classification of Transportation Packaging and Dry Spent Fuel Storage Systems</i>, as follows.</p> <ul style="list-style-type: none"> • Category A - Critical to Safe Operation SSCs in this category include those whose failure or malfunction could directly result in a condition adverse to public health and safety. Important-to-safety SSCs in this category are classified as safety class as defined in DOE Order 6430.1A, with the additional requirements therein. • Category B - Major Impact on Safety SSCs in this category include those whose failure or malfunction could result in a condition adversely affecting collocated worker health and safety. Note that from the definition of Category C, Category B is understood to include events that could significantly damage the MCO without severe impact to public health and safety. SSCs in this category are classified as safety significant as defined in DOE-STD-3009-94. • Category C - Minor Impact on Safety SSCs whose failure or malfunction would not significantly reduce the MCO 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 safety class 3 (nonsafety) as defined in DOE Order 6430.1A. 	DOE 6430.1A HNF-PRO-516	<p>The MCO Topical Report (HNF-SD-SNF-SARR-005), Chapter 3.0, Section 3.1.10.3, contains the NRC equivalency "important-to-safety" classifications for major MCO SSCs in accordance with the definitions proposed in HNF-SD-SNF-DB-005, Rev. 3, submitted for DOE approval July 1998.</p> <p>The Quality Assurance Program requirements applicable for MCO fabrication are documented in the MCO Fabrication Specification (HNF-S-0453), including specific requirements relative to procurement of safety-class, safety-significant, and general service materials.</p>

Item No.	Additional NRC requirements as identified in HNF-SD-SNF-DB-005, Rev. 3	Related DOE/Project Hanford references	Assessment of compliance for MCO
11 (cont.)	<p>Worker safety issues are addressed through DOE Orders 6430.1A and 5480.23 and DOE-STD-3009-94.</p> <p>(Reference: 10 CFR 72.3, "Definitions," 10 CFR 72.106, "Controlled area of an ISFSI or MRS," 10 CFR 72.122, "Overall requirements," and 10 CFR 72.140(b), "Quality assurance requirements," NUREG/CR-6407, <i>Classification of Transportation Packaging and Dry Spent Fuel Storage Systems</i>)</p>		
12	<p>Incorporate the requirements of Regulatory Guide 8.8 into the design of the MCO. Example considerations include the process and configuration associated with welding the MCO cover cap in place, inservice inspection and maintenance, and other MCO-related activities that could impact ALARA.</p> <p>(Reference: Regulatory Guide 8.8, <i>Information Relative to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable</i>, Rev. 3)</p>	<p>10 CFR 835</p> <p>DOE 5480.11</p> <p>DOE 6430.1A</p>	<p>Compliance with Regulatory Guide 8.8 is required in Section 4.13 of the MCO Performance Specification (HNF-S-0426). The MCO Design Report (HNF-SD-SNF-DR-003), Section 3.13, discusses the application of ALARA principles in the MCO design.</p> <p>The design of the MCO shield plug reflects numerous features selected to achieve ALARA. The thickness of the stainless steel plug has been maximized as much as possible to minimize the dose at the top of the shield plug. Examples of compliance with Regulatory Guide 8.8 include passageways through the shield plug with 90 degree angles to prevent streaming in accordance with Section 2.b.4 and Section 2.b.3 and shield plug appurtenances designed to be actuated via both remote-operated and long-handled tooling to reduce dose by means of distance.</p> <p>The selection of the mechanical closure was based, in part, on study results that included detailed dose/manhour estimates for the specific operation steps involved in closing the MCO and cask.</p>

Item No.	Additional NRC requirements as identified in HNF-SD-SNF-DB-005, Rev. 3	Related DOE/Project Hanford references	Assessment of compliance for MCO
13	<p>Include the following radiological exposure criteria considerations for the design and safety analyses of the MCO and its interfaces with the CSB and CVDF since the MCO provides a containment/confinement function for the K Basins SNF.</p> <ul style="list-style-type: none"> • Ensure the 10 CFR 20.1301 hourly dose limit of 0.002 rem to the public for any unrestricted area from external sources during normal operations and anticipated occurrences is met. • Ensure during normal operations and anticipated occurrences for any real individual of the public that the following 10 CFR 72.104 annual dose equivalent criteria are met: 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ. <p>(References: 10 CFR 20, "Standards for Protection Against Radiation," 10 CFR 72.104, "Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS," and 10 CFR 72.126, "Criteria for radiological protection")</p>	<p>10 CFR 835 DOE 5400.5 DOE 5480.11 DOE 5480.23 DOE 6430.1A DOE-STD-3009-94 HSRCM-1 HNF-PRO-704</p>	<p>The MCO shield plug, combined with the cask or other shielding barrier (MHM or CSB weld station), provide the necessary shielding to meet the annual radiological exposure criteria. Development of estimated annual dose-equivalent exposures is facility dependent and is being performed by the CVDF and CSB subprojects. Results of these dose estimates, demonstrating compliance with exposure criteria, will be documented in the CVDF and CSB facility FSARs.</p>
	<p>"Refer to WHC-SD-SNF-DB-004, <i>Spent Nuclear Fuel Project Seismic Design Criteria, NRC Equivalency Evaluation Report</i>, Rev. 2, for details on seismic criteria.</p> <p>"Refer to WHC-SD-SNF-DB-002, <i>Spent Nuclear Fuel Project Path Forward Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities</i>, Table 5.c, Sections 72.24 and 72.122, for related information.</p> <p>ALARA = as low as reasonably achievable. ASME = American Society of Mechanical Engineers. DOE = U.S. Department of Energy. DOE-RL = U.S. Department of Energy, Richland Operations, Office. ES&H = environment, safety, and health. FSAR = final safety analysis report. MCO = multi-canister overpack. MHM = multi-canister overpack handling machine. NRC = U.S. Nuclear Regulatory Commission. SAR = safety analysis report. SNF = spent nuclear fuel. SSC = structure, system, or component. WHC = Westinghouse Hanford Company.</p>		

Item No.	Additional NRC requirements as identified in HNF-SD-SNF-DB-005, Rev. 3	Related DOE/Project Hanford references	Assessment of compliance for MCO
References:			
	10 CFR 20, 1995, "Standards for Protection Against Radiation," <i>Code of Federal Regulations</i> .		
	10 CFR 21, 1995, "Reporting of Defects and Noncompliance," <i>Code of Federal Regulations</i> .		
	10 CFR 50, 1995, "Domestic Licensing of Production and Utilization Facilities," <i>Code of Federal Regulations</i> .		
	10 CFR 72, 1995, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," <i>Code of Federal Regulations</i> .		
	10 CFR 830, 1995, "Nuclear Safety Management," <i>Code of Federal Regulations</i> .		
	10 CFR 835, 1995, "Occupational Radiation Protection," <i>Code of Federal Regulations</i> .		
	AP 2-15, "Operations: Occurrence Reporting and Processing of Operation Information," <i>Spent Nuclear Fuel Project Administrative Procedures</i> , DE&S Hanford, Incorporated, Richland, Washington.		
	DOE Order 5000.3B, 1993, <i>Occurrence Reporting and Processing of Operations Information</i> , U.S. Department of Energy, Washington, D.C.		
	DOE Order 5400.5, 1993, <i>Radiation Protection of the Public and the Environment</i> , U.S. Department of Energy, Washington, D.C.		
	DOE Order 5480.11, 1988, <i>Radiation Protection for Occupational Workers</i> , U.S. Department of Energy, Washington, D.C.		
	DOE Order 5480.23, 1992, <i>Nuclear Safety Analysis Reports</i> , U.S. Department of Energy, Washington, D.C.		
	DOE Order 5480.24, 1992, <i>Nuclear Criticality Safety</i> , U.S. Department of Energy, Washington, D.C.		
	DOE Order 5480.28, 1993, <i>Natural Phenomena Hazards Mitigation</i> , U.S. Department of Energy, Washington, D.C.		
	DOE Order 6430.1A, 1989, <i>General Design Criteria</i> , U.S. Department of Energy, Washington, D.C.		
	DOE-STD-3009-94, 1994, <i>Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports</i> , U.S. Department of Energy, Washington, D.C.		
	HNF-PRO-060, <i>Reporting Occurrences and Processing Operations Information</i> , Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-PRO-097, <i>Engineering Design and Evaluation</i> , Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-PRO-298, <i>Nonconforming Item Reporting and Control</i> , Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-PRO-334, <i>Criticality Safety General Requirements</i> , Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-PRO-537, <i>Criticality Safety Control of Fissionable Material</i> , Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-PRO-540, <i>Criticality Prevention Specifications</i> , Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-PRO-544, <i>Criticality Plant Configuration Control</i> , Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-PRO-545, <i>Fissionable Material Packaging and Transportation</i> , Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-PRO-704, <i>Hazard and Accident Analysis Process</i> , Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-3553, <i>Spent Nuclear Fuel Project Final Safety Analysis Report</i> , Annex A, "Canister Storage Building Final Safety Analysis Report," and Annex B, "Cold Vacuum Drying Facility Final Safety Analysis Report," Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-S-0425, 1998, <i>Specification for Spent Nuclear Fuel Canister Storage Building</i> , Rev. 3, Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-S-0426, 2000, <i>Performance Specification for Spent Nuclear Fuel Multi-Canister Overpack</i> , Rev. 6, Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-S-0453, 1998, <i>Specification for Multi-Canister Overpack Fabrication</i> , Rev. 2F, Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-SD-SNF-CSE-005, 1997, <i>Criticality Safety Evaluation Report for Spent Nuclear Fuel Processing and Storage Facilities</i> , Rev. 3, Fluor Daniel Hanford, Incorporated, Richland, Washington.		
	HNF-SD-SNF-DB-005, 1998, <i>Spent Nuclear Fuel Project Multi-Canister Overpack, Additional NRC Requirements</i> , Rev. 3, Fluor Daniel Hanford, Incorporated, Richland, Washington.		

Item No.	Additional NRC requirements as identified in HNF-SD-SNF-DB-005, Rev. 3	Related DOE/Project Hanford references	Assessment of compliance for MCO
References (continued):			
HNF-SD-SNF-DR-003, 2000, <i>Multi-Canister Overpack Design Report</i> , Rev. 3, Fluor Hanford, Incorporated, Richland, Washington.			
HNF-SD-SNF-DRD-002, 1998, <i>Cold Vacuum Drying Facility Design Requirements</i> , Rev. 3, Fluor Daniel Hanford, Incorporated, Richland, Washington.			
HNF-SD-SNF-SARR-005, 1998, <i>Multi-Canister Overpack Topical Report</i> , Rev. 1, Fluor Daniel Hanford, Incorporated, Richland, Washington.			
HNF-SP-1228, 1997, <i>Quality Assurance Program Implementation Plan for Nuclear Facilities</i> , Rev. 0, Fluor Daniel Hanford, Richland, Washington.			
HSRCM-1, 1994, <i>Hanford Site Radiological Control Manual</i> , Rev. 2, Westinghouse Hanford Company, Richland, Washington.			
IEN 95-29, 1995, <i>Overview of Design and Fabrication Activities for Metal Components Used in Spent Fuel Dry Storage Systems</i> , U.S. Nuclear Regulatory Commission, Washington, D.C.			
NUREG-0612, 1980, <i>Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36</i> , U.S. Nuclear Regulatory Commission, Washington, D.C.			
NUREG-0800, 1981, <i>Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants</i> , SRP 9.1.2, "Spent Fuel Storage," Rev. 3, U.S. Nuclear Regulatory Commission, Washington, D.C.			
NUREGCR-3854, 1984, <i>Fabrication Criteria for Shipping Containers</i> , U.S. Nuclear Regulatory Commission, Washington, D.C.			
NUREGCR-4461, 1986, <i>Tornado Climatology of the Contiguous United States</i> , U.S. Nuclear Regulatory Commission, Washington, D.C.			
NUREGCR-6407, 1996, <i>Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety</i> , U.S. Nuclear Regulatory Commission, Washington, D.C.			
Regulatory Guide 1.26, 1976, <i>Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants</i> , Rev. 3, U.S. Nuclear Regulatory Commission, Richland, Washington.			
Regulatory Guide 1.60, 1973, <i>Design Response Spectra for Seismic Design of Nuclear Power Plants</i> , Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.			
Regulatory Guide 1.61, 1973, <i>Damping Values for Seismic Design of Nuclear Power Plants</i> , Rev. 0, U.S. Nuclear Regulatory Commission, Washington, D.C.			
Regulatory Guide 1.76, 1974, <i>Design Basis Tornado for Nuclear Power Plants</i> , U.S. Nuclear Regulatory Commission, Washington, D.C.			
Regulatory Guide 1.84, 1993, <i>Design and Fabrication Code Case Acceptability ASME Section III, Division I</i> , Rev. 29, U.S. Nuclear Regulatory Commission, Washington, D.C.			
Regulatory Guide 1.85, 1993, <i>Materials Code Case Acceptability ASME Section III, Division I</i> , Rev. 29, U.S. Nuclear Regulatory Commission, Richland, Washington.			
Regulatory Guide 3.48, 1989, <i>Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)</i> , Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.			
Regulatory Guide 8.8, 1978, <i>Information Relative to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable</i> , Rev. 3, U.S. Nuclear Regulatory Commission, Washington, D.C.			
SECY-93-087, 1993, <i>Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs</i> , U.S. Nuclear Regulatory Commission, Washington, D.C.			
SNPP-1, <i>Policies, Charters, and Assignments</i> , Section 1.4, Attachment A, "Spent Nuclear Fuel Project Quality Assurance Program Plan," Rev. 1, DE&S Hanford, Incorporated, Richland, Washington.			
WHC-SD-SNF-DB-002, 1996, <i>Spent Nuclear Fuel Project Path Forward, Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities</i> , Rev. 2, Westinghouse Hanford Company, Richland, Washington.			
WHC-SD-SNF-DB-004, 1996, <i>Spent Nuclear Fuel Project Seismic Design Criteria, NRC Equivalency Evaluation Report</i> , Rev. 2, Fluor Daniel Hanford, Incorporated, Richland, Washington.			

Item No.	Additional NRC requirements as identified in HNF-SD-SNF-DB-005, Rev. 3	Related DOE/Project Hanford references	Assessment of compliance for MCO
References (continued):			
WHC-SD-SNF-DB-009, 1996, <i>Canister Storage Building Natural Phenomena Hazards</i> , Rev. 4, Westinghouse Hanford Company, Richland, Washington.			
WHC-SD-SNF-DB-010, 1996, <i>Cold Vacuum Drying System Natural Phenomena Hazards</i> , Rev. 1, Westinghouse Hanford Company, Richland, Washington.			
WHC-SP-1131, 1996, <i>Westinghouse Hanford Company Quality Assurance Program and Implementation Plan</i> , Rev. 1, Westinghouse Hanford Company, Richland, Washington.			

2.0 PRINCIPAL DESIGN CRITERIA

2.1 SPENT FUEL TO BE STORED

Chapter 1.0 of this Topical Report provided background information related to the multi-canister overpack (MCO) and a description of the MCO and its general characteristics. When fully loaded, each MCO will house five or six baskets, each basket containing prescribed amounts of spent nuclear fuel (SNF) and incidental fuel corrosion products that have been physically or chemically bound to the fuel and retrieved with the fuel from the K Basins. The following fuel element materials will be contained in any one of those baskets at any given time. A description of the fuel types and characteristics will enable the reader to view health and safety concerns in the proper perspective as additional details pertaining to the MCO are presented in this report. The following paragraphs and tables contain physical descriptions of the fuel currently in the K Basins, give a fuel burnup summary, and provide details of the chemical and radionuclide inventories of SNF. The information presented also is useful in supporting the chapters in this report that discuss radiological protection, shielding, and criticality.

N Reactor fuel assemblies consist of two concentric tubes made of uranium metal coextruded into Zircaloy-2 cladding. The two basic types of fuel assemblies are differentiated by their uranium enrichment. Mark IV fuel assemblies have a pre-irradiation enrichment of 0.947% ^{235}U in both tubes and an average uranium weight of 22.7 kg. The Mark IV assemblies have an outside diameter of 6.15 cm (2.42 in.) and lengths of 44.2 cm, 58.9 cm, 62.5 cm, or 66.3 cm (17.4, 23.2, 24.6, or 26.1 in.). A small amount of fuel with 0.71 wt% ^{235}U enrichment was designated as Mark IVB fuel. The Mark IVB fuel has the same dimensions and weights as the 66.3-cm- (26.1-in.-) long Mark IV fuel. Table 2-1 contains a detailed physical description of the fuel.

Mark IA fuel assemblies have a pre-irradiation enrichment of 1.25% ^{235}U in the outer tube and 0.947% ^{235}U in the inner tube. The Mark IA assemblies have an average uranium weight of 16.3 kg. Mark IA fuel assemblies have an outside diameter of 6.10 cm (2.40 in.) and lengths of 37.8 cm, 49.8 cm, 53.1 cm, or 66.3 cm (14.9, 19.6, 20.9, or 26.1 in.). The long Mark IA assemblies (26.1 in.) will be loaded in a Mark IV basket. Bounding analyses for criticality (using two scrap baskets) have been performed and have yielded acceptable results (see Chapter 6.0).

Throughout the MCO Topical Report, there are references to the "fuel" that is contained within the MCO. In almost all cases, the fuel being referred to is the N Reactor fuel and is either Mark IA or Mark IV fuel. There is, however, a small amount of single pass reactor (SPR) fuel currently in the K Basins. The SPR fuel consists of a machined uranium core that was sealed within, and metallurgically bonded, to an aluminum can. There were three basic designs for SPR fuel elements: solid core cylinders, hollow core cylinders, and tubes. SPR fuel will be loaded into a single MCO containing Mark IA fuel and scrap baskets. The fuel baskets will be modified to contain an insert to permit the SPR fuel to be stacked 2-3 elements high to ensure efficient packing densities and permit all the fuel to fit into one MCO. The fuel will be processed using the currently designed fuel retrieval system equipment, including the cleaning machine. The single

Table 2-1. 105-N Reactor Fuel Assembly Description.

	Mark IV	Mark IA
Pre-irradiation enrichment of ^{235}U	0.947% enriched	1.25-0.947% enriched
Type-length code ^a	E S A C	E M T F
Length (cm)	66.3 62.5 58.9 44.2	66.3 ^b 53.1 49.8 37.8
Element diameter (cm)		
1. Outer of outer	6.15	6.10
2. Inner of outer	4.32	4.50
3. Outer of inner	3.25	3.18
4. Inner of inner	1.22	1.11
Cladding mass (kg)		
1. Outer element	1.09 1.04 0.99 0.79	1.07 0.88 0.83 0.66
2. Inner element	0.55 0.52 0.50 0.40	0.66 0.54 0.51 0.40
Mass of uranium in outer element (kg)		
1. 0.947% ^{235}U	16.0 15.0 14.1 10.5	
2. 1.25% ^{235}U		13.8 11.1 10.4 7.85
Mass of uranium in inner element (kg) 0.947% ^{235}U	7.48 7.03 6.62 4.94	6.84 5.49 5.12 3.90
Weighted average of uranium in element (kg)	22.7	16.3
Ratio of Zircaloy-2 to uranium (kg/MTU)	70.0 70.8 71.6 77.1	83.8 85.5 86.3 90.4
Weighted average (kg/MTU)	70.3	85.7
% of total elements ^c	63	37
% of length type of each fuel	78 10 7 5	0.03 ^b 87 10 3
Displacement volume (L/MTU)	67	67

^aLetter codes differentiate the different lengths of the Mark IV or Mark IA fuel elements (e.g., a type "E" element is 66.3 cm long).

^bThere are only 12 of the 66.3-cm Mark IA assemblies.

^cPercent based on the total element count.

MTU = metric ton of uranium.

MCO will be initially loaded with the fuel in the K East Basin and then transferred to the K West Basin to complete the loading process. The MCO will then be processed using the currently designed cold vacuum drying equipment. Following processing at the Cold Vacuum Drying Facility (CVDF), it will be transferred to the Canister Storage Building (CSB) for interim storage.

The SPR fuel has not been documented as extensively as the N Reactor fuel, making it very difficult to treat the fuels similarly in the report. It is apparent that the SPR fuel is bounded by the N Reactor fuel that is discussed (Praga 1998a). Although most of the SPR fuel (approximately 3 metric ton of uranium [MTU] of the 3.38 MTU at the Hanford Site) was irradiated to a higher ^{240}Pu fraction (26%), the highest burnup N Reactor fuel has a higher concentration of plutonium, americium, and other radionuclides that dominate dose consequences in a release. In addition, the SPR fuel has experienced a much longer decay period than the N Reactor fuel, so the N Reactor fuel has a larger concentration of gamma-emitting isotopes; therefore shielding requirements are set by N Reactor fuel.

The condition of the fuel also is a consideration. Although photographs of the SPR fuel indicate the presence of a large amount of debris and sludge, the uranium corrosion that is evident in some of the N Reactor fuel is not observed for the SPR fuel. As a final comparison, an MCO fully loaded with the longest length Mark IV N Reactor fuel would contain over 6 MTU of fuel, while the entire inventory of all SPR fuel is only 3.38 MTU. The SPR fuel will not be loaded into MCOs with N Reactor fuel but will, instead, be in its own MCO. These factors form the basis for the belief that the MCO containing the SPR fuel is bounded by the MCOs containing the N Reactor fuel. The N Reactor fuel also is bounding for criticality as discussed in Chapter 6.0. Because of the above, no special or separate consideration is given to the SPR fuel in the MCO Topical Report.

Exposure level, or fuel burnup, and time since discharge determine the radionuclide content of a fuel assembly or group of assemblies. The K Basin inventory of N Reactor fuel is composed of elements that experienced a range of exposure levels and were discharged from the reactor between January 1971 and April 1987. The exposure levels ranged from unburned fuel (0 MWd) to approximately 6,000 MWd per MTU. Burnup is related to the weight fraction of the isotope ^{240}Pu that exists within the total quantity of plutonium in a particular fuel element. The ^{240}Pu content is commonly used to indicate fuel burnup and ranges from approximately 0 wt% ^{240}Pu up to 16.72 wt% ^{240}Pu for the N Reactor fuel in inventory.

Accountability records have been used as the basis for estimating the radionuclide content of N Reactor fuel. The accountability record run data, dated November 17, 1994, includes discharge date, fuel type, ^{240}Pu content, and other information for 497 groups of fuel elements (the groups are also known as keys). Each group, or key, includes elements of the same type, with the same burnup, that were discharged from the reactor at the same time. The mass of uranium associated with each key varies from 2.37×10^{-3} MTU to 67.4 MTU. The accountability database that forms the inventory basis, and which takes precedence over this Topical Report in fuel inventory matters, is shown in Appendix A of HNF-SD-SNF-TI-009, *105-K Basin Material Design Basis Feed Description for Spent Nuclear Fuel Project Facilities* (Praga 1998a). Tables 2-2 and 2-3 present summaries of the fuel burnup listed in the accountability database for Mark IV and Mark IA fuel.

Table 2-2. N Reactor Mark IV Fuel Burnup Summary.

Percent ^{240}Pu (range)	Mass (MTU)	Percent of total uranium mass
≤ 5	166.94	11.35
>5 - 7	125.25	8.51
>7 - 9	0.059761	0.00
>9 - 11	62.988	4.28
>11 - 13	270.56	18.39
>13 - 15	714.01	48.54
>15	131.29	8.92
Total	1471.1	100

MTU = metric ton of uranium.

Table 2-3. N Reactor Mark IA Fuel Burnup Summary.

Percent ^{240}Pu (range)	Mass (MTU)	Percent of total uranium mass
≤ 5	36.124	5.75
>5 - 7	3.3729	0.54
>7 - 9	0	0.00
>9 - 11	68.008	10.83
>11 - 13	118.59	18.88
>13 - 15	401.88	64.00
>15	0	0.00
Total	627.98	100

MTU = metric ton of uranium.

The chemical content of the fuel on a pre-irradiation basis is derived by applying the reported pre-irradiation concentration range to the total uranium (2,100,000 kg), zirconium (145,000 kg), and brazing filling (3,000 kg) in N Reactor fuel. In the post-irradiation fuel, a small percentage of the uranium will have been fissioned or converted to plutonium, and some of the other constituents will have been activated by neutron bombardment; this is particularly true for the boron and hafnium, which have relatively high thermal neutron capture cross sections. Table 2-4 lists the chemical inventory (kg) of N Reactor fuel currently stored in the K Basins (Praga 1998a)

Tables 2-5, 2-6, and 2-7 contain the radionuclide inventory of the N Reactor fuel in storage at the K Basins. Table 2-5 contains an inventory estimate for the combined total for both basins. Table 2-6 contains the shielding basis inventory per metric ton of uranium, and Table 2-7 contains the safety radiological inventory and decay power per metric ton of uranium (Praga 1998a). The ⁸⁵Kr data provided in each table assumes this gaseous nuclide has not been released from the fuel.

2.2 DESIGN CRITERIA FOR ENVIRONMENTAL CONDITIONS AND NATURAL PHENOMENA HAZARDS

Natural phenomena hazard (NPH) loads applicable to the MCOs are specified in this section. The U.S. Department of Energy (DOE) NPH requirements are based on DOE Order 5480.28, *Natural Phenomena Hazards Mitigation*; and supporting standards, DOE-STD-1020-94, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*; DOE-STD-1022-94, *Natural Phenomena Hazards Characterization Criteria*; and DOE-STD-1023-95, *Natural Phenomena Hazards Assessment Criteria*. DOE regulatory policy for SNF Project activities also requires a level of nuclear safety comparable to that of U.S. Nuclear Regulatory Commission (NRC)-licensed facilities (Grumbly 1995). The NRC NPH requirements are based on 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). Shipping will be prohibited during severe weather conditions that could place the cask-MCO at risk.

DOE Order 5480.28 requires that each structure, system, and component (SSC) be assigned to one of five performance categories based on safety class and hazard category. Each performance category has an associated NPH goal that serves as a measure of the level of protection against potential natural phenomena. The CSB has been given an interim designation as a hazard category 2 facility (Kummerer 1995; Sellers 1996). The CVDF is also designated as a hazard category 2 facility. With these classifications, safety-class SSCs within these facilities will be designated performance category 3 SSCs.

Table 2-4. Chemical Inventory of N Reactor Fuel Currently Stored in the K Basins.

Element	Uranium alloy 601 (kg)	Zircaloy-2 cladding (kg)	Brazing filler (kg)	Totals (kg)*
Al	1,480 - 1,900	11.1	0.411	1,700
B	0.530	0.074	0.00142	0.605
Be	21.0	--	142	163
C	769 - 1,550	40.7	1.42	1,200
Cd	0.530	0.074	0.00142	0.605
Co	--	1.48	0.0567	1.54
Cr	137	74 - 222	1.42 - 4.26	288
Cu	158	7.40	0.170	166
Fe	632 - 843	104 - 296	1.70 - 5.96	941
H	4.22	3.70	0.142	8.06
Hf	--	29.6	0.567	30.2
Mg	52.7	2.96	0.170	55.8
Mn	52.7	7.40	0.170	60.3
Mo	--	7.40	0.142	7.54
N	158	11.8	0.567	170
Na	--	2.96	0.0567	3.02
Ni	211	44.4 - 118	0.851 - 2.27	294
O	--	--	6.53	6.53
Pb	--	14.8	0.369	15.2
Si	261	14.8	0.709	277
Sn	--	1,780 - 2,520	32.3 - 48.2	2,190
Ti	--	7.40	0.142	7.54
V	--	7.40	0.142	7.54
W	--	7.40	0.284	7.68
Zr	753	145,000	2,780	148,000
Actinides				
U	2,100,000	0.518	0.0113	2,100,000
Np	81.1	--	--	81.1
Pu	4,120	--	--	4,120
Am	109	--	--	109
Cm	0.018	--	--	0.018
Fission products				
Se	12.0	--	--	12.0
Sr	152	--	--	152
Tc	170	--	--	170
Pd	133	--	--	133
Kr	77.1	--	--	77.1
I	46.1	--	--	46.1
Cs	4.65	--	--	4.65
Pm	0.50	--	--	0.50
Sm	176	--	--	176
Xe	1,040	--	--	1,040

*For the values with a range, the midpoint of the range is used in calculating chemical inventory (includes preburnup and postburnup values).

Table 2-5. Radionuclide Inventory of the Combined K Basins. (2 sheets)

Isotope	Activity (Ci)	Mass (kg)	Heat generation (W)	Isotope	Activity (Ci)	Mass (kg)	Heat generation (W)
Fission and activation products							
H-3	3.66e+04	3.79e-03	1.23e+00	Sn-123	1.74e-05	2.12e-12	5.42e-08
C-14	6.93e+02	1.55e-01	2.02e-01	Sn-126	1.56e+02	5.50e+00	4.78e-02
Fe-55	1.84e+03	7.36e-04	6.12e-02	Sb-124	3.03e-18	1.73e-25	4.01e-20
Co-60	3.96e+03	3.50e-03	6.09e+01	Sb-125	3.35e+04	3.24e-02	1.05e+02
Ni-59	4.10e+01	5.41e-01	1.63e-03	Sb-126	2.18e+01	2.61e-07	3.96e-01
Ni-63	4.49e+03	7.28e-02	4.54e-01	Sb-126m	1.56e+02	1.99e-09	2.00e+00
Se-79	8.62e+01	1.24e+00	2.66e-02	Te-123m	2.76e-11	3.11e-18	3.98e-14
Kr-85	5.90e+05	1.50e+00	8.85e+02	Te-125m	8.18e+03	4.54e-04	6.89e+00
Sr-89	0.00	0.00	0.00	Te-127	9.49e-07	3.60e-16	1.28e-09
Sr-90	1.01e+07	7.40e+01	1.17e+04	Te-127m	9.69e-07	1.03e-13	4.81e-10
Y-90	1.01e+07	1.86e-02	5.60e+04	Te-129	0.00	0.00	0.00
Y-91	2.23e-14	9.09e-22	8.00e-17	Te-129m	0.00	0.00	0.00
Zr-93	4.00e+02	1.59e+02	4.61e-02	I-129	6.37e+00	3.61e+01	2.99e-03
Zr-95	1.69e-12	7.85e-20	8.49e-15	Cs-134	1.59e+04	1.23e-02	1.62e+02
Nb-93m	2.47e+02	8.74e-04	4.42e-02	Cs-135	7.75e+01	6.73e+01	2.59e-02
Nb-95	3.74e-12	9.55e-20	1.79e-14	Cs-137	1.32e+07	1.52e+02	1.33e+04
Nb-95m	1.25e-14	3.29e-23	1.65e-17	Ba-137m	1.25e+07	2.32e-05	4.89e+04
Tc-99	2.88e+03	1.70e+02	1.44e+00	Ce-141	0.00	0.00	0.00
Ru-103	0.00	0.00	0.00	Ce-144	9.14e+02	2.86e-04	6.04e-01
Ru-106	1.82e+03	5.44e-04	1.08e-01	Pr-143	0.00	0.00	0.00
Rh-103m	0.00	0.00	0.00	Pr-144	9.03e+02	1.20e-08	6.62e+00
Rh-106	1.82e+03	5.11e-10	1.74e+01	Pr-144m	1.10e+01	6.06e-11	3.69e-03
Pd-107	1.63e+01	3.17e+01	8.94e-04	Pm-147	4.62e+05	4.98e-01	1.70e+02
Ag-110	2.84e-04	6.81e-17	2.13e-09	Pm-148	0.00	0.00	0.00
Ag-110 m	2.14e-02	4.50e-09	3.55e-04	Pm-148 m	0.00	0.00	0.00
Cd-113m	3.55e+03	1.64e-02	3.90e+00	Sm-151	1.76e+05	6.69e+00	2.06e+01
Cd-115m	0.00	0.00	0.00	Eu-152	9.45e+02	5.46e-03	4.28e+00
In-113m	2.14e-07	1.28e-17	4.95e-10	Eu-154	1.07e+05	3.96e-01	9.64e+02
Sn-113	2.14e-07	2.13e-14	3.55e-11	Eu-155	2.21e+04	4.75e-02	1.60e+01
Sn-119m	2.97e-01	6.62e-08	1.53e-04	Gd-153	1.28e-04	3.63e-11	1.15e-07
Sn-121m	7.96e+01	1.35e-03	8.36e-02	Tb-160	2.77e-15	2.45e-22	2.21e-17
Fission and activation product totals					4.74e+07	7.07e+02	1.32e+05

Table 2-5. Radionuclide Inventory of the Combined K Basins. (2 sheets)

Isotope	Activity (Ci)	Mass (kg)	Heat generation (W)	Isotope	Activity (Ci)	Mass (kg)	Heat generation (W)
Actinides							
U-234	8.74e+02	1.40e+02	2.47e+01	Pu-241	6.68e+06	6.48e+01	2.07e+02
U-235	3.37e+01	1.56e+04	9.14e-01	Pu-242	5.49e+01	1.44e+01	1.60e+00
U-236	1.27e+02	1.96e+03	3.40e+00	Am-241	3.75e+05	1.09e+02	1.23e+04
U-238	6.96e+02	2.07e+06	1.73e+01	Am-242	1.95e+02	2.41e-07	1.62e+00
Np-237	5.72e+01	8.11e+01	1.65e+00	Am-242 m	1.96e+02	2.02e-02	2.99e-01
Pu-238	1.11e+05	6.48e+00	3.63e+03		1.20e+02	6.02e-01	3.81e+00
Pu-239	2.18e+05	3.51e+03	6.72e+03	Am-243	1.62e+02	4.90e-05	5.86e+00
Pu-240	1.19e+05	5.22e+02	3.71e+03	Cm-242	1.44e+03	1.78e-02	4.95e+01
				Cm-244			
Actinide totals					7.51e+06	2.09e+06	2.66e+04

Note: 06/02/97 RADNUC2A run for combined K Basins, with results decayed to 05/31/98. Total fuel mass in the combined basins, 2.10e+03 MTU. Total fuel activity in the combined basins, 5.50e+07 Ci. Total fuel heat generation in the combined basins, 1.61e+05 W. Plutonium-239 and plutonium-240 have been adjusted to ensure consistency with data previously reported by Safeguards.

MTU = metric ton of uranium.

Table 2-6. Shielding Design Basis (Based on Mark IV Fuel at 16 Percent Plutonium-240 Aged 13.5 Years). (2 sheets)

Isotope	Activity (Ci/MTU)	Heat generation (W/MTU)	Isotope	Activity (Ci/MTU)	Heat generation (W/MTU)
Fission and activation products					
H-3	4.03e+01	1.36e-03	Sn-123	4.08e-09	1.26e-11
C-14	5.27e-01	1.54e-04	Sn-126	1.22e-01	3.75e-05
Fe-55	5.23e+00	1.75e-04	Sb-124	0.00	0.00
Co-60	6.27e+00	9.64e-02	Sb-125	0.00	0.00
Ni-59	3.03e-02	1.21e-06	Sb-126	1.71e-02	3.10e-04
Ni-63	3.53e+00	3.57e-04	Sb-126m	1.22e-01	1.57e-03
Se-79	6.23e-02	1.93e-05	Te-123m	1.19e-13	1.72e-16
Kr-85	6.23e+02	9.33e-01	Te-125m	0.00	0.00
Sr-89	0.00	0.00	Te-127	1.16e-10	1.57e-13
Sr-90	8.19e+03	9.50e+00	Te-127m	1.19e-10	5.88e-14
Y-90	8.19e+03	4.53e+01	Te-129	0.00	0.00
Y-91	0.00	0.00	Te-129m	0.00	0.00
Zr-93	2.83e-01	3.25e-05	I-129	4.88e-03	2.29e-06
Zr-95	4.20e-18	2.12e-20	Cs-134	1.07e+02	1.10e+00
Nb-93m	1.38e-01	2.47e-05	Cs-135	5.77e-02	1.93e-05
Nb-95	9.33e-18	4.48e-20	Cs-137	1.13e+04	1.14e+01
Nb-95m	3.12e-20	4.12e-23	Ba-137m	1.07e+04	4.18e+01
Tc-99	2.08e+00	1.05e-03	Ce-141	0.00	0.00
Ru-103	0.00	0.00	Ce-144	1.75e+00	1.15e-03
Ru-106	9.38e+00	5.59e-04	Pr-143	0.00	0.00
Rh-103m	0.00	0.00	Pr-144	1.73e+00	1.26e-02
Rh-106	9.38e+00	8.97e-02	Pr-144m	2.10e-02	7.05e-06
Pd-107	1.44e-02	7.94e-07	Pm-147	1.06e+03	3.88e-01
Ag-110	4.15e-06	3.11e-11	Pm-148	0.00	0.00
Ag-110m	3.12e-04	5.18e-06	Pm-148m	0.00	0.00
Cd-113m	3.96e+00	4.35e-03	Sm-151	1.08e+02	1.26e-02
Cd-115m	0.00	0.00	Eu-152	1.22e+00	5.50e-03
In-113m	2.59e-11	5.98e-14	Eu-154	2.02e+02	1.81e+00
Sn-113	2.59e-11	4.30e-15	Eu-155	3.42e+01	2.48e-02
Sn-119m	0.00	0.00	Gd-153	3.92e-06	3.53e-09
Sn-121m	0.00	0.00	Tb-160	2.22e-19	1.78e-21
Fission and activation product totals				4.05e+04	1.12e+02

Table 2-6. Shielding Design Basis (Based on Mark IV Fuel at
16 Percent Plutonium-240 Aged 13.5 Years). (2 sheets)

Isotope	Activity (Ci/MTU)	Heat generation (W/MTU)	Isotope	Activity (Ci/MTU)	Heat generation (W/MTU)
Actinides					
U-234	3.92e-01	1.11e-02	Pu-241	9.42e+03	2.92e-01
U-235	1.31e-02	3.54e-04	Pu-242	7.46e-02	2.16e-03
U-236	7.12e-02	1.89e-03	Am-241	2.92e+02	9.57e+00
U-238	3.35e-01	8.35e-03	Am-242	3.19e-01	2.66e-03
Np-237	4.42e-02	1.27e-03	Am-242m	3.21e-01	4.91e-04
Pu-238	1.28e+02	4.16e+00	Am-243	2.22e-01	7.02e-03
Pu-239	1.68e+02	5.14e+00	Cm-242	2.65e-01	9.59e-03
Pu-240	1.28e+02	3.90e+00	Cm-244	4.62e+00	1.59e-01
Actinide totals				1.01e+04	2.33e+01

Note: 06/02/97 RADNUC2A run for shielding design basis (Mark IV fuel at 16% ²⁴⁰Pu). Results decayed to 05/31/98. Total mass of fuel in this run, 2.60e+00 MTU. Total activity of fuel in this run, 1.32e+05 Ci. Total heat generation from the fuel in this run, 3.55e+02 W.

MTU = metric ton of uranium.

Table 2-7. Safety/Regulatory Assessment Design Basis (Based on Mark IV Fuel at 16.72 Percent Plutonium-240). (2 sheets)

Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)	Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)
Fission and activation products					
H-3	2.61e+01	8.77e-04	Sn-123	1.72e-16	5.35e-19
C-14	5.53e-01	1.62e-04	Sn-126	1.29e-01	3.97e-05
Fe-55	5.41e-01	1.80e-05	Sb-124	0.00	0.00
Co-60	2.09e+00	3.23e-02	Sb-125	0.00	0.00
Ni-59	3.18e-02	1.26e-06	Sb-126	1.81e-02	3.28e-04
Ni-63	3.47e+00	3.51e-04	Sb-126m	1.29e-01	1.66e-03
Se-79	6.54e-02	2.02e-05	Te-123m	1.50e-21	2.18e-24
Kr-85	3.70e+02	5.53e-01	Te-125m	0.00	0.00
Sr-89	0.00	0.00	Te-127	2.12e-19	2.85e-22
Sr-90	6.93e+03	8.03e+00	Te-127m	2.16e-19	1.07e-22
Y-90	6.93e+03	3.84e+01	Te-129	0.00	0.00
Y-91	0.00	0.00	Te-129m	0.00	0.00
Zr-93	2.95e-01	3.40e-05	I-129	5.16e-03	2.41e-06
Zr-95	0.00	0.00	Cs-134	6.47e+00	6.59e-02
Nb-93m	1.93e-01	3.45e-05	Cs-135	6.04e-02	2.02e-05
Nb-95	0.00	0.00	Cs-137	9.66e+03	9.78e+00
Nb-95m	0.00	0.00	Ba-137m	9.14e+03	3.59e+01
Tc-99	2.19e+00	1.10e-03	Ce-141	0.00	0.00
Ru-103	0.00	0.00	Ce-144	7.91e-04	5.23e-07
Ru-106	2.56e-02	1.52e-06	Pr-143	0.00	0.00
Rh-103m	0.00	0.00	Pr-144	7.82e-04	5.73e-06
Rh-106	2.56e-02	2.45e-04	Pr-144m	9.48e-06	3.18e-09
Pd-107	1.56e-02	8.56e-07	Pm-147	1.09e+02	4.02e-02
Ag-110	7.17e-10	5.38e-15	Pm-148	0.00	0.00
Ag-110m	5.39e-08	8.94e-10	Pm-148m	0.00	0.00
Cd-113m	2.78e+00	3.06e-03	Sm-151	1.02e+02	1.19e-02
Cd-115m	0.00	0.00	Eu-152	8.45e-01	3.81e-03
In-113m	1.36e-19	3.13e-22	Eu-154	1.13e+02	1.01e+00
Sn-113	1.36e-19	2.26e-23	Eu-155	1.06e+01	7.65e-03
Sn-119m	6.14e-08	3.16e-11	Gd-153	5.19e-10	4.67e-13
Sn-121m	6.27e-02	6.59e-05	Tb-160	0.00	0.00
Fission and activation product totals				3.34e+04	9.38e+01

Table 2-7. Safety/Regulatory Assessment Design Basis (Based on Mark IV Fuel at 16.72 Percent Plutonium-240). (2 sheets)

Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)	Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)
Actinides					
U-234	3.84e-01	1.09e-02	Pu-241	6.82e+03	2.11e-01
U-235	1.27e-02	3.44e-04	Pu-242	8.71e-02	2.55e-03
U-236	7.16e-02	1.91e-03	Am-241	4.34e+02	1.42e+01
U-238	3.31e-01	8.23e-03	Am-242	3.71e-01	3.08e-03
Np-237	4.66e-02	1.34e-03	Am-242m	3.72e-01	5.71e-04
Pu-238	1.33e+02	4.34e+00	Am-243	2.78e-01	8.82e-03
Pu-239	1.73e+02	5.28e+00	Cm-242	3.08e-01	1.11e-02
Pu-240	1.37e+02	4.19e+00	Cm-244	4.47e+00	1.54e-01
Actinide totals				7.70e+03	2.85e+01

Note: 06/02/97 RADNUC2A run for the safety basis (Mark IV fuel at 16.72% ²⁴⁰Pu). Results decayed to 05/31/98. Total mass of the fuel in this run, 1.16e+01 MTU. Total activity of the fuel in this run, 4.77e+05 Ci. Total heat generation of the fuel in this run, 1.42e+03 W.

MTU = metric ton of uranium.

An MCO must maintain its structural integrity so as to perform the following functions for all credible events:

- Limit the release path for particulate material to the penetrations associated with the rupture disk and vent path (facility specific) (see Section 2.3.2)
- Limit the path for oxygen ingress to these same two paths (common to all facilities [see Chapter 11.0])
- Maintain the geometry and physical controls assumed in the criticality safety evaluation report (common to all facilities, see Chapter 6.0)
- Prevent water ingress (see Chapter 11.0)
- Prevent gaseous release (see Chapter 11.0).

The MCOs will perform the above-listed functions at the CVDF, to be located in the 100 K Area, and at the CSB, to be located in the 200 East Area. The facility's SSCs required to ensure these MCO functions are maintained must be designed to performance category 3 requirements. The following discussions address the NPH loadings at both of these locations on the Hanford Site.

2.2.1 Tornado and Wind Loadings

According to the DOE requirements and guidance provided in DOE Order 5480.28 and DOE-STD-1020-94, the Hanford Site does not have a design basis tornado. However, to implement NRC nuclear safety equivalency as identified in HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements* (Garvin 1997b), a design basis tornado must be identified and included in the designs for the CVDF and the CSB. Tornado protection for the MCO will be provided by the facilities in which it resides. Tornado protection for the MCO while it is in the K Basins is not required as this aspect of NRC nuclear safety equivalency was not extended by HNF-SD-SNF-DB-003 (Garvin 1997b) to K Basins activities. The three selected equivalency issues applied to K Basins activities relate to two quality assurance considerations and criticality control (Garvin 1997b).

2.2.1.1 Forces on Structures. While the MCO is located within the CVDF and CSB, the associated structure will shield the MCO such that it will not experience an extreme wind or tornado wind loading or impact from extreme wind-driven missiles.

2.2.1.2 Tornado-Generated Missiles. Protecting the MCO from tornado-generated missiles is facility specific. The following options may be applied:

- Provide a structure or barrier that does not allow for missile penetration that would put the MCO at risk

- Show by analysis that none of the credible or NRC-postulated missiles will penetrate the MCO or the shipping cask-MCO combination
- Show that the risk of a significant radiological release caused by missile impact is acceptably low (acceptance criteria for $<1.0 \times 10^{-6}$ events/yr using conservative methods with a dose acceptance criterion of 5 rem (Tallman 1996a, 1996b) (see Section 3.1.3).

For the first and second options, the load combinations must be in accordance with NUREG-0800, *Standard Review Plan*, Section 3.3.2, "Tornado Loading" (NRC 1981). It is the third option that has been applied to the CVDF and the CSB.

If extreme fog, ice, or adverse snow conditions exist, the cask and MCO shall not be transported. If the ambient air temperature is 4 °C (40 °F) or less (as determined by the Hanford Site weather station), personnel shall drive the shipping route to ensure there is no appreciable ice or snow on the road immediately prior to shipment. If it is determined that adequate traction exists, the shipment may be made. If visibility is less than 200 m (0.125 mi) because of fog, the shipment shall not be made. Driving the route between the K Basins and the CVDF will not be required.

2.2.2 Design Basis Flood

Within the CVDF and the CSB, the MCO will always be protected against the design basis runoff flood level for the probable maximum precipitation (PMP). This will be accomplished through facility design and by providing sufficient runoff capacity. The MCO will be protected from the river probable maximum flood (PMF) by locating the CVDF and the CSB above the PMF level. Specific performance category 3 values for the PMP and PMF are provided for the CSB in WHC-SD-SNF-DB-009, *Canister Storage Building Natural Phenomena Hazards* (Tallman 1996a), and for the CVDF in WHC-SD-SNF-DB-010, *Cold Vacuum Drying System Natural Phenomena Hazards* (Tallman 1996b). These performance category 3 NPH values are applied to the safety-class MCO and to those SSCs that function to protect the MCO. With this protection provided for the MCO by the facilities, there is no need to perform analyses of the loading that the PMP or PMF might have on an MCO. The flood protection provided by each facility is described in the facility-specific safety analysis report (SAR).

2.2.3 Seismic-System Analyses

The CSB seismic design criteria is 0.35 g for horizontal and 0.23 g for vertical accelerations. The design basis for the MCO includes 100 g for a side drop and 35 g for an end drop. The CSB seismic design criterion is less than one one-hundredth of the MCOs designed acceleration force capacity. To approach the design basis accident (DBA) allowable forces on an MCO during an earthquake the magnitude of the seismic design criterion, the CSB facility would, at a minimum, have to multiply the earthquake acceleration forces by a factor of 100. It is reasonable to assume

that the CSB is not capable of putting a multiplier of 100 on the MCO's payload. The MCO design basis drop accelerations conservatively bound any CSB imposed accelerations caused by an earthquake (see Section 3.1.3).

2.2.4 Snow and Ice Loadings

In the CVDF and the CSB, the MCO will always be protected against snow and ice loadings by facility designs that ensure that neither will make contact with the MCO. As such, snow and ice will not stress the MCO either by temperature transients or extremes, or by dead weight loading. During transport, the cask and cask lid will protect the MCO.

2.2.5 Combined Load Criteria

The confinement features of the MCOs are designed and fabricated to meet the load combination criteria in the *Boiler and Pressure Vessel Code* (ASME 1998), Section III, Subsection NB (see Sections 3.1 and 3.2).

2.2.6 Baseline Load Criteria

This section describes the criteria selected to provide the baseline loadings, including temperatures, which when met will ensure the mechanical and structural integrity of the MCO. The loadings and criteria are taken from HNF-S-0426, *Performance Specification for the Spent Nuclear Fuel Multi-Canister Overpack* (Goldmann 2000b), and are presented in the following paragraphs.

The purpose of the MCO is to confine, contain, and maintain SNF in a critically safe array. Use of a criticality control exclusion void is discussed in Chapters 3.0 and 6.0.

2.2.6.1 Performance Criteria Loadings. The MCO consists of a shell, a shield plug, a baseplate with lower radial support plates, a center insert criticality or basket support tube, a full complement of baskets designed to hold particular specified fuels, process tubes, and incidental equipment. These components, as an assembly, are subjected to the following performance criteria loadings.

- The MCO shall maintain fuel elements or fragments thereof in a critically safe array throughout its 40-year design life and during and after being subjected to DBAs.
- The MCO, when fitted with an active rupture disk, shall relieve internal pressure in excess of 150 lb/in² gauge.
- MCO handling features and equipment shall be capable of holding the maximum fully loaded weight of the component in question. In the case of the total MCO package,

this includes the heaviest loaded fuel arrangement, plus water, plus margin, bringing the total handling load to 10,900 kg.

- The MCO shall be capable of performing its function when exposed to atmospheric temperatures ranging from -33 °C to 46 °C (-27 °F to 115 °F) and relative humidity ranging from 5% to 100%.

2.2.6.2 Design Loadings. The following loadings apply to the MCO design.

- The thermal source term internal to the MCO from the confined SNF is an average of 403 W and a maximum of 776 W.
- The radioactive source term for worst-case shielding and dose consequences consists of 270 Mark IV fuel elements of 0.95% ²³⁵U irradiated to 16% ²⁴⁰Pu. Activity for these fuels is found in Tables 2-6 and 2-7 and are taken from HNF-SD-SNF-TI-009 (Praga 1998a). Radiation effects have been considered for polymers to ensure seals can withstand radiation exposure (see Section 5.2).
- The internal design pressure is 150 lb/in² gauge before the cover cap is in place and 450 lb/in² gauge after the cover cap is installed.
- The design temperature is 132 °C (270 °F). For design features requiring temperatures at the various MCO life cycle stages, the following MCO shell temperatures are appropriate:
 - K Basin loading, 6 °C to 38 °C (43 °F to 100 °F)
 - Cold vacuum drying, 10 °C to 50 °C (50 °F to 122 °F)
 - Transportation, -17 °C to 73 °C (2 °F to 163 °F)
- Fuel temperature during staging and storage may be up to 157 °C (315 °F).
- The design temperature differential between any portions of the MCO confinement boundary materials (i.e., the shell and shield plug) is 100 °C (180 °F).
- In addition to the design load limits, certain fuel and MCO shell limitations are associated with facility management.
 - During MCO handling machine (MHM) handling, staging, and storage, the MCO temperatures are allowed to go to 132 °C (270 °F). The facility may limit temperatures further if they choose. The fuel temperature may be up to 157 °C (315 °F) during normal steady-state conditions.

2.2.6.2.1 Design Temperature. The MCO design basis is at 132 °C and at 150 lb/in² using the mechanical closure and 450 lb/in² using the welded cover cap configuration. The design basis acceleration levels used for MCO drop analyses are 35 g for an end drop and 101 g for a

side drop. The design bases were arrived at in conjunction with the decision to code stamp the MCO. During processing and storage, the MCO may experience unplanned transients during which the temperatures of some components may exceed the 132 °C. Increasing the temperatures on the stainless steels decreases the ASME Boiler and Pressure Vessel Code stress intensity allowables (ASME 1998). For example, increasing the temperature to 375 °C from 132 °C on 304 or 304L stainless steel material decreases the stress allowable by approximately 20% (ASME 1995). The loss in code stress allowables because of elevated temperatures begins at 149 °C. So at 149 °C the material is as good as the material when at the 132 °C design temperature and may be treated as the same. ASME Boiler and Pressure Vessel Code stress allowables are less than the actual material properties. The use of the MCO stainless steel materials at temperatures up to 427 °C is acceptable from a material degradation standpoint. The continuous operation of MCOs in air at this temperature will certainly discolor the air contacted surface areas but will not appreciably affect the bulk material properties. In normal service, the MCO's shell and basket materials are well below the 132 °C design basis.

Table 4-1 summarizes analytical results for extraordinary conditions in which the temperature of the MCOs is maximized. The design basis for the MCO is 132 °C, and the MCO is not expected to go beyond that temperature in normal service. However, if the MCO components are allowed to heat up beyond 132 °C, an evaluation will have to be made. The MCO is quite capable of satisfactory service beyond 149 °C.

Scenario 1

Table 4-1 includes various extreme temperature scenarios that the MCO components could possibly experience in their lifetimes. In Table 4-1, the maximum steady state temperature of the MCO in CSB storage is 131 °C for the fuel, 122 °C for the MCO internal gas, and 108 °C for the pressure boundary materials (i.e., MCO wall). The basket material temperature will range from a peak of approximately 126 °C, or about 5 °C below the peak fuel temperature, to 119 °C, or about the fuel temperature at the outer edge of the basket. The bulk average basket material temperature will be approximately equal to that of the bulk gas temperature, or 122 °C. This scenario is based on steady-state design day ambient temperature, which is an incredible situation.

Scenario 2

Table 4-1 also indicates that for an MCO in the MHM it takes 125 to 167 hours (depending on the assumed starting conditions) for a maximum radiolytical decay heat powered MCO shell to reach 132 °C with the MHM's forced (MHM extract fan) convection off. By the end of the transient in this hypothetical scenario, the MCO basket components will average 142 °C, or equal to the MCO internal gas temperature. The hottest fuel temperature will be 153 °C when the MCO shell reaches 132 °C. The MCO component temperatures for the above scenarios demonstrate the typical thermal gradient that will be established inside the MCO under steady-state conditions. The level of decay and chemical heat generated within the fuel and the thermal resistance between the center of the basket and the MCO shell determines the delta temperature, which will be established under steady-state conditions. Except in the transient condition that would exist when an MCO is initially exposed to an environment that is hotter than itself, the

driving force represented by the decay and chemical heat will eventually force the fuel at the center of the basket to be the hottest component within the MCO.

Temperature conditions of the scenarios above are neither normal nor expected. However, if they should occur and create component temperatures near and possibly beyond the design basis, mitigation of the event may be needed and is discussed below.

- CSB storage tube

Temperatures of the MCO's pressure shell and baskets in the CSB storage tubes will not exceed 132 °C in the long term. This conclusion is based on the above discussion of Table 4-1. Getting the MCO material temperatures above 132 °C will require more MCO driving heat for removal or severely restricting the CSB vault's ambient cooling air flow.

- Long-term breakdown of MHM extract fan

Loss of the MHM extract fan when an MCO is in transit will be noticed because the MHM has indicators that show the fan is down. In the event the MCO components were allowed to go beyond the 132 °C design temperature with a broken MHM fan system, the MCO shell and basket materials normally would be first cooled to 132 °C or less before the MHM is further moved or the MCO handled. Before moving an MCO with component temperatures higher than 132 °C, an evaluation would have to be made that would include the actual materials' properties at the temperature above 132 °C, payload weight, potential drops, and expected results as related to material deformation. As most of the design analysis was done using code stress allowables, the actual properties of the MCO components will be available when the MCO pressure boundaries and baskets are put into service. The code allowables are always less than the actual properties. If the basket and MCO shell temperatures are less than 149 °C, the MCO is still adequate using code allowables for the 132 °C design basis. At temperatures beyond 149 °C, the above-mentioned evaluation will have to be made.

The hottest fuel is at 153 °C and, with the known delta T of 5 °C, the basket material maximum temperature is about 148 °C. As the basket material temperature is not beyond 149 °C, the structural adequacy of the basket is assured. At temperatures above 149 °C, an evaluation will have to be made as previously mentioned.

During possible events in which the MCO's cooling has been interrupted or the ambient cooling conditions artificially constrained, the baskets inside the MCO will experience the higher temperatures. These accidents will most likely occur when the MCO is stationary or in process. The safety-class baskets are designed to accommodate the forces generated from the minimum 35 g drop at the design temperature of 132 °C. This excess strength capability is more than adequate to offset the reduction in material properties due to the material's temperature possibly extending beyond 149 °C.

Seals used in the MCO are designed for use at temperatures greatly exceeding the 132 °C design temperatures for the MCO. The Helicoflex seal is constructed of silver, stainless steel, and alloy metal materials. The manufacturer rates the Helicoflex seal at a 370 °C continuous temperature. The port cover plate and appliance seals are made of Inconel² and are plated with silver or gold. These noble metal-plated port seals are fully capable to favorably withstand temperatures of up to 375 °C. Relatively speaking, these seals are more capable than the stainless steel materials of the baskets and the pressure boundary shield plug and shell.

2.2.6.2.2 Design Pressure. The discussion in this section considers the MCO without the welded cover cap installed and is based on MCO nominal design dimensions and material properties taken from certified material test reports of testing and training MCOs recently fabricated by Oregon Iron Works in Klackamas, Oregon. Joints created by welding are assumed to match base metal properties.

The MCO, without a welded cover cap attached, consists of the shell assembly with the mechanical closure assembly installed in the shell assembly's collar. The shell assembly includes the collar, shell wall, and the bottom cap. This welded assembly is hydrostatically pressure tested at the fabrication shop to 450 lb/in² x 1.25 or 562.5 lb/in². During this hydrostatic test, the collar's buttress threads carry the internally generated axial pressure loading of the hydrostatic test.

The shell assembly has various zones of stress. These zones are broken out by type of stress and section thickness and are each discussed individually here.

1. Above the Helicoflex seal face at the buttress thread root

Above the seal face in the collar, the weakest point is the thread relief root. This area is considered to be 0.380 in. in thickness. The collar material's yield is 38,000 lb/in² with an ultimate of 81,500 lb/in². As this area is above the seal face, with no cover cap this section is only loaded axially, so approximately 1,130,000 lb of pull is needed to put the entire section into tension yield at an average 38,000 lb/in² loading of the section. This corresponds to an internal pressure of approximately 2,680 lb/in² on the pressure area of the threaded plug. The capability of the plug to retain the pressure at the seal face is assumed in this case.

2. Just below the seal face in the collar

Below the seal face on the collar the section is 1.155 in. thick. This section is in axial and circumferential (hoop) stress loading. To produce a uniform 38,000 lb/in² stress in the section and begin yielding, the internal pressure would have to be approximately 3,415 lb/in².

²Inconel is a trademark of INCO Alloys International, Incorporated.

The collar material has an ultimate of 81,500 lb/in², which is more than double the initial yield value of 38,000 lb/in², so there is additional conservatism available.

3. The 0.5-in. plate shell wall

The 0.5-in.-thick shell wall section of the shell assembly has a 46,000 lb/in² yield and 87,400 lb/in² ultimate. At approximately 1,790 lb/in² internal pressure, the shell would reach the point of yield and start stretching.

4. The bottom plate

The bottom plate is 2 in. thick except for the center sump, which is 2.5 in. in diameter and the bottom plate thickness is 1.13 in.

The mechanical closure assembly is the same as the previously analyzed 450 lb/in² design. New calculations have derated the assembly components to the 150 lb/in² level because of the change of the shell assembly's collar material from FXM-19 (Nitronic 50) to 304/304L dual-certified stainless steel. The 18 set screws roughly match the section area of the thread root section and are of higher property material. The lifting and locking ring properties are also higher than those of the collar material. Any limiting deformation in the closure area due to MCO internal pressure will occur in the collar buttress thread area and not in the mechanical closure assembly, excluding the port appliances and cover plates.

The existing 150 lb/in² design for the assembled mechanical closure MCO has a safety factor such that using the actual material properties and the ASME Boiler and Pressure Vessel Code calculations (ASME 1998) would allow a rating of 225 lb/in². Using primary stress, actual material properties, and yield, the design is good up to 340 lb/in² with no permanent deformation. So the MCO with the mechanical closure is stable at an internal loading of 340 lb/in². When the internal pressure increases, the thread root becomes more stressed and more of the thread root area is brought into higher tension stress conditions. Thus, going beyond the 340 lb/in² pressure range, the thread root of the collar will start to yield in tension at the inner surface of the collar. The offset force loading on the buttress threads creates a bending moment at the thread root, which initiates the yielding. Then, at a much higher pressure, the whole section thickness of 0.38 in. will be at or beyond initial yield.

As a side note, a proposed vendor for the production MCO built and pressure tested a prototype MCO with 304/304L dual-certified collar material. The MCO was pressure tested to 562.5 lb/in² (450 lb/in² x 1.25) and neither main seal leaking nor permanent deformation of the MCO collar was found. This activity will be repeated with the actual MCO production fabrication vendor as part of the fabrication contract.

Because of the previous experience in pressure testing and because of stress allowables and criteria used from subsection NB of ASME Boiler and Pressure Vessel Code, Section III (ASME 1998), in the MCO design, the MCO has an embedded safety factor of 3 to 4. With the

design pressure of the shield plug mechanical closure system at 150 lb/in², there remains adequate margin in the event of accidental 155 lb/in² combustion pressure.

After the initial compression of sealing, the Helicoflex seal is capable of elastically rebounding about 0.012 in. while still maintaining the seal. To get the Helicoflex seal to begin to leak, the collar thread root portion of the section must be stretched to something less than the 0.012 in. needed for complete seal relaxation. Estimating that the stretch actually occurs in a 0.25 in. along the MCO axis, a 4% stretch of the collar would be needed. As stated, the pressure has to reach approximately 2,680 lb/in² to put the whole 0.38-in.-thick collar section into yield. Before the pressure reaches the 2,680 lb/in² level, the shell would begin to stretch at approximately 1,790 lb/in². The expectation is that a leak will begin at the Helicoflex seal as the pressure increases above 340 lb/in². Where the actual "stretch" or deformation in the materials of the mechanical closure system occurs in order to produce a "leak" past the Helicoflex has not been determined.

2.2.6.3 Design Basis Accident Loadings. The following DBA loadings are required. They must meet the Service Level D requirements of the Boiler and Pressure Vessel Code (ASME 1998), Section III, Subsection NB, for safety-class items or lower class items required to prevent failure of the safety-class items.

- The MCO shall withstand a DBA fire on the outside of a closed cask. This fire will reach a temperature of 800 °C (1,472 °F) with an emissivity coefficient of at least 0.9 for not less than 30 minutes. The fire will raise the MCO shell temperature to 122 °C (252 °F) for 180 minutes after the fire (see Section 3.1.3).
- While at the CSB, the MCO shall maintain confinement and subcriticality during a design basis earthquake having a zero period, 5% damped, horizontal ground acceleration of 0.35 g. Response spectra for other damping values are provided in WHC-SD-W236A-TI-002, *Probabilistic Seismic Hazard Analysis, DOE Hanford Site, Washington* (Geomatrix 1996) (see Section 3.1.3).
- While at the CVDF, the MCO shall maintain confinement and subcriticality during a design basis earthquake having a zero period, 5% damped, horizontal ground acceleration of 0.26 g. Response spectra for other damping values are provided in WHC-SD-W236A-TI-002 (Geomatrix 1996) (see Section 3.1.3).
- The MCO shall survive accelerations created or bounded by the drops analyzed in Chapter 3.0 and described in HNF-S-0426 (Goldmann 2000b) while maintaining confinement and subcriticality (the temperature range for the drops is 25 °C to 132 °C [75 °F to 270 °F] and the pressure range is 0.0 to 450 lb/in² gauge). Additional detail on the drop analyses is presented in Chapter 3.0.

A loaded basket subject to criticality controls required in HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for Spent Nuclear Fuel Processing and Storage Facilities* (Kessler 2000), such as the Mark IA, must not have or be subjected to stress levels higher than

the applicable service level requirements allowed in the Boiler and Pressure Vessel Code (ASME 1998), Subsection NG; any exceptions to this will be noted and defined; handling loads for both loaded and unloaded baskets will include a safety factor of three on material yield and five on material ultimate strength of the baskets.

2.3 SAFETY PROTECTION SYSTEMS

2.3.1 General

The activities associated with the MCO and its contents have been reviewed from initial loading of the MCO through interim storage. The following sections discuss the protection afforded the MCO throughout its 40-year life cycle. The various forms of protection range from the physical structures housing the MCO to instrumentation associated with the various facilities.

2.3.2 Protection by Confinement Barriers and Systems

Within the shipping cask, CVDF, and CSB, an MCO is not the sole barrier against radiological release. This is different from most facilities licensed under 10 CFR 72. In those facilities, the cask (the MCO equivalent but with integral shielding) is the only radionuclide barrier beyond the fuel cladding.

For the SNF Project, radionuclide releases from an MCO are analyzed by considering facility-specific mitigation features, siting considerations affecting atmospheric dispersion, and the location of the onsite and offsite receptors. Credible oxygen ingress and vent paths are by design limited to those discussed in Sections 2.2 and 11.3.2. As a radionuclide barrier, the MCO must not fail under any credible condition. However, maintaining this feature has the following facility-specific considerations.

- Within the CSB, impact absorbers must be provided in the storage tubes as required to prevent failure of a dropped MCO and underlying MCO (see Section 2.2.6.3, fourth bullet).
- Within the CSB, safety-class features must be provided to prevent the MHM from shearing an MCO by moving while the MCO is in an intermediate position in a tube (Garvin 1997a).
- In the CSB, the structural integrity of the tube system, inclusive of the upper floor penetration and lower base plate support, is taken credit for as a safety-class item to prevent common mode failure of the MCOs and to maintain configuration for criticality prevention.

- Maintaining the MCO in a thermally safe situation for prevention of fuel fire is also a safety function.

The following are MCO-specific features designed to achieve the lowest practical level of radioactive release from an MCO to the facility during normal and off-normal conditions.

- The MCO is provided with an internal bank of four high-efficiency particulate air (HEPA) filters. As these process filters cannot be tested after MCO assembly, no credit is taken for their presence in accident analyses.
- The MCO is designed to remain intact for all credible events (see Chapter 11.0).
- Prior to CSB transport, the MCO is provided with an active rupture disk designed to relieve at 150 lb/in² gauge. With this protection, should the MCO overpressurize (caused, for example, by water heatup), the radionuclide release will be limited by relief through the rupture disk passage.

During shipment from the K Basins to the CVDF, the MCO filtered process port must be open, allowing the MCO to vent into the cask. During shipment to the CSB, the MCO is sealed and the rupture disk is inactive.

The means by which the MCO is protected against off-normal operations and external loadings are transport cask and facility-specific considerations and are addressed in the SAR for packaging (Smith 2000) and each facility's SAR (Garvin 1997a, Pili-Vincens 1998, WHC 1997). Each facility will be responsible for providing operators and equipment to address such considerations.

2.3.3 Protection by Equipment and Instrumentation Selection

2.3.3.1 Pressure Relief Equipment. When filled with water, the MCO is provided with a safety-class rupture disk to relieve internal pressures in excess of the normal operating pressure (Goldmann 1996a). The rupture disk meets the following design criteria (Goldmann 2000b).

- The rupture disk must relieve MCO internal pressures at a pressure greater than 150 lb/in² in the temperature range of 10 °C to 132 °C (50 °F to 270 °F). The rupture disk will operate at 32 °F. The rupture disk is required to operate in accordance with the ASME Boiler and Pressure Vessel Code (ASME 1998), Sections NB-7611 and NB-7612(a) and (b)2. The range for determining the burst pressure versus temperature curve is 32 °F to 270 °F. The specified burst temperature is 270 °F with the burst range being 150 lb/in² ±5% in accordance with the applicable section of the ASME Boiler and Pressure Vessel Code (ASME 1998). The disks will therefore rupture between 142.5 lb/in² and 157.5 lb/in². The ASME Boiler and Pressure Vessel Code includes margin to accommodate a burst below the

157.7 lb/in² level. The actual pressure variation of $\pm 5\%$ does not significantly affect the release calculation in the event of rupture disk burst.

- The rupture disk must perform the above functions after being subject to the DBAs.
- The rupture disk must perform the above functions in a humid environment (up to 100% humidity for short-term only) and a corrosive atmosphere.
- The rupture disk must perform the above functions without preventive maintenance.
- The rupture disk must be designed for contact assembly and replacement, if necessary, using appropriate handling equipment as required for working on the MCO. Design consideration shall be given to DOE Order 6430.1A, *General Design Criteria*.
- The exposed surfaces shall be such as to facilitate their decontamination.

2.3.3.2 High-Efficiency Particulate Air Filter. The MCO is provided with an internal bank of four HEPA filters. The function of the HEPA filters is to retain particulate inside the MCO during drying (Goldmann 1996a).

2.3.3.3 Instrumentation. Currently, no instrumentation is provided with the MCO. However, addition of a pressure-readout system for monitoring purposes is likely. Details of the monitoring instrumentation will be included in future updates of this Topical Report.

For operating systems requiring pressure indication, necessary instrumentation will be attached to the process ports of the shield plug. If necessary for the function being performed, the systems also provide temperature, flow, and constituent measurements. Measurement of the MCO environment for operational and safety purposes shall be conducted and recorded by the instrumentation connected to the process ports. As a minimum, the following functions will be monitored, as required, in succession:

- Gas purge system at the K Basins (includes cask)
- Water drain, vacuum drying, and gas purge systems at the CVDF
- Sampling, purge systems at the CSB.

2.3.4 Nuclear Criticality Safety

The criticality safety evaluation and controls used for design of the MCO and its components and systems that are important to safety are detailed in Chapter 6.0 of this report. The analyses performed consider both the intact fuel and the scrap basket contents. The criticality analyses show acceptable results with zero, one, or two baskets of scrap when loaded as described in Chapter 6.0.

2.3.5 Radiological Protection

The MCO design in combination with other equipment and facilities, incorporates shielding features to reduce radiation dose levels from nuclear material confinement, handling, and storage. Internal filters are provided to retain radioactive particulates inside the MCO and reduce potential for contamination. Stainless steel material is used and irregular surfaces or shapes are minimized to the extent possible to enhance surface decontamination. Hardware and connections are designed for ease of use to minimize "hands-on" exposure time for normal packaging and handling activities.

Handling of the MCO and operations involving the MCO will be done in accordance with facility-specific procedures developed to ensure safe operations and as-low-as-reasonably-achievable (ALARA) radiation exposure. Facility-specific operations will address the requirements in Project Hanford radiation protection policies and procedures and applicable portions of NRC Regulatory Guide 8.8, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable* (NRC 1978). Facility procedures implement requirements of Title 10, *Code of Federal Regulations*, Part 835, "Occupational Radiation Protection" (10 CFR 835), and are consistent with the elements of an acceptable occupational ALARA program as established in Implementation Guide G-10 CFR 835/B2, Revision 2, "Occupational ALARA Program." Appropriate measures for radiological protection and control will be based on the nature of work activities, operations, and hazards present at each specific facility. The aforementioned Project Hanford radiation protection policies and procedures and NRC Regulatory Guide 8.8 (NRC 1978) provide a comprehensive list of measures for managing and controlling radiation exposure to employees at levels that are ALARA. Radiological protection and ALARA programs will be an integral part of work planning, operations, and maintenance activities (see also Chapter 10.0).

2.3.6 Hydrogen Combustion Protection

Many of the conditions contributing to the hydrogen combustion scenarios are specific to the various facilities and operations to which the MCO and its contents will be subjected. Accordingly, details regarding the handling of these issues will be contained in the facility-specific SAR chapters.

2.4 DECOMMISSIONING CONSIDERATIONS

At some time, the MCOs stored in the CSB will have to be dispositioned. It is not now apparent exactly how the MCOs will be handled or what requirements will be in place in the future. It is logical to assume that either the MCO will be determined to be transportable in some manner and therefore be shipped to an alternate storage location or repository, or it will be determined that the MCO should be opened and the contents repackaged for transport or onsite dispositioning. If the MCO is opened and the fuel removed, the MCO itself would be

decontaminated to the appropriate level and disposed of or recycled. This activity will probably fall under the responsibility of the CSB, the facility most likely to be opening the MCO.

The MCO design lends itself to decontamination. The outer shell is generally smooth, allowing surface decontamination efforts to proceed readily. The cover cap also provides the capability for sampling the MCO to ensure internal environments are acceptable before opening the MCO. Similarly, if the MCO is to be opened and repackaged, the mechanical closure will enable such activities to take place without the need for extensive weld removal or unacceptable worker exposure. The internals of the MCO also provide readily accessible individual containers of fuel or scrap that will facilitate the dispositioning processes. All of these features will assist in the decontamination and decommissioning of the MCO.

Hydride ignition concerns are not presently analyzed for as they pertain to decontamination and decommissioning activity and are therefore not authorized by this Topical Report. Also, removal of the cover cap for decontamination and decommissioning is not authorized after the MCO is put into interim storage.

3.0 STRUCTURAL EVALUATION OF THE MULTI-CANISTER OVERPACK

3.1 STRUCTURAL DESIGN

The structural evaluation of the multi-canister overpack (MCO) reflects the MCO's dual design pressure. The design pressure is 450 lb/in² for an MCO with a welded cover cap in place, whereas the design pressure is 150 lb/in² for an MCO with the shield plug sealed to the shell but without the cover cap welded in place.

3.1.1 Multi-Canister Overpack

The MCO is a stainless steel vessel used to stabilize and store the spent nuclear fuel (SNF) currently stored in the Hanford K Basins. The SNF will be placed in an appropriate fuel basket and stacked inside the MCO. The fuel stabilization process includes (1) transporting the loaded MCO to the Cold Vacuum Drying Facility (CVDF), (2) cold vacuum drying the fuel within the MCO, (3) transporting the MCO loaded with dried fuel to the Canister Storage Building (CSB), (4) welding an additional pressure boundary cap on the MCO, and (5) placing the MCO inside a storage tube for 40 years of storage.

HNF-S-0426, *Performance Specification for Spent Nuclear Fuel Multi-Canister Overpack* (Goldmann 2000b), requires that the MCO be designed in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code*, Section III, Subsection NB (ASME 1998). The design loadings for pressure, temperature, and mechanical loads must meet the stress intensity limits for the stress categories noted in Design Condition Figure NB-3221-1 of the *Boiler and Pressure Vessel Code* (ASME 1998). In addition, the limits for design loadings shall meet the requirements of the appropriate subsections of Section NCA-2142.4 (ASME 1998).

The Design Agent, Parsons Infrastructure and Technology, Richland, Washington, was responsible for analysis and design that meets the requirements of the MCO Performance Specification (Goldmann 2000b).

3.1.2 Code Requirements

The MCO is designed in accordance with applicable sections of DOE Order 6430.1A, *General Design Criteria*. Safety-class and safety-significant components providing fuel confinement and criticality control have been designed and will be constructed in accordance with the ASME *Boiler and Pressure Vessel Code*, Section III, Subsection NB, "Rules for Construction of Nuclear Power Plant Components" (ASME 1998), under the component safety group as guided by the NUREG/CR 3854, *Fabrication Criteria for Shipping Containers* (NRC 1984). The U.S. Nuclear Regulatory Commission (NRC) positions on ASME Code Section III cases, as provided in Regulatory Guide 1.84, *Design and Fabrication Code Case Acceptability, ASME Section III, Division 1* (NRC 1993a), and Regulatory Guide 1.85, *Materials Code Case*

Acceptability, ASME Section III, Division I (NRC 1993b), were reviewed for their applicability to the MCO design.

3.1.3 Multi-Canister Overpack Design Criteria

The MCO has been designed to meet the following criteria under normal operating conditions (Goldmann 1998):

- ASME Section III Code stamp. required
- Design pressure for shell assembly and cover cap. 450 lb/in² at 132 °C (270 °F)
- Design pressure for shield plug closure assembly. 150 lb/in² gauge at 132 °C (270 °F)
- Process operating pressure
 - Full vacuum internal, with 60 lb/in² gauge external pressure, at 75 °C (167 °F)
 - Full vacuum internal, with 0.0 lb/in² gauge external pressure, up to 132 °C (270 °F)
 - 75 lb/in² gauge internal, with 0.0 lb/in² gauge external pressure, up to 132 °C (270 °F)
- Differential thermal expansion. 0.65 in. in axial direction between the basket stack and the MCO shell, maintaining basket nesting and engagement of the top basket with the shield plug
- Maximum design radial temperature gradient. 100 °C (180 °F) between the outside of the MCO shell and the center of the MCO shield plug and 5 °C (9 °F) within the MCO shell wall.

The 100 °C differential temperature selected as the design basis temperature was agreed to by all users of the MCO in advance of the design. The differential temperature applies in both directions (i.e., the exterior of the MCO may be hotter or cooler). The hotter exterior, as analyzed in the design report, is the most conservative of the two heat flow cases, so the analysis was performed that way. A hotter exterior will result in a tendency for the collar to move radially away from the shield plug. An analysis was conducted with the maximum differential temperature set at 100 °C to determine the temperature distribution. Results showed that under the differential temperatures set for design basis conditions, the gradient in the shell was about 5 °C. In normal service, the MCO pressure boundary will never experience differential temperature conditions beyond the 100 °C differential in the design criteria. Axial gradients may occur because MCOs may contact non-heat producing structures that transmit heat away from the ends

of the MCOs. However, no credit was taken for this heat transfer, and the structural effects are bounded by the radially oriented thermal analysis of the MCO.

All safety-class items shall maintain SNF confinement and subcriticality during and after the design basis accidents (DBAs) listed below. All safety-significant items whose failure could result in the failure of such safety-class items also shall be designed to withstand the DBAs listed below. The following DBA loadings are required to meet the service level D requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB (ASME 1998), for these items.

- **Design basis fire** (temperatures resulting from exposure to a design basis accidental fire on the outside of the transportation cask [from 10 CFR 71.73 (3)]). This design basis fire shall result in exposure of the outside of the transportation cask to a heat flux not less than that of a radiation environment of 800 °C (1,472 °F) with an emissivity coefficient of at least 0.9 for not less than 30 minutes. For purposes of calculation, the surface absorptivity must be either that value the shipping cask may be expected to possess if exposed to a fire or 0.8, whichever is greater. (This fire will raise the MCO shell temperature to 122 °C [252 °F] for 180 minutes after the fire. The 132 °C [270 °F] design temperature bounds the temperature associated with this fire.)
- **Design basis earthquake.** The horizontal ground acceleration for the design basis earthquake at the CSB is 0.35 g. Although this will be amplified because of the position of the MCOs within the facility, the DBA drop accelerations bound any imposed earthquake accelerations from the K Basins, CSB, or CVDF.
- **Design basis tornado.** The CSB and CVDF incorporate features that prevent or mitigate radionuclide releases from MCOs caused by tornadoes or wind-driven missiles (as determined necessary by probabilistic risk assessments).
- **Design basis hydrogen deflagration.** The MCO shall maintain confinement during a design basis hydrogen deflagration event (ASME service level D event) beginning at atmospheric pressure inside the MCO at 75 °C (167 °F).
- **Design basis drops.** The design basis drops have been determined to create accelerations that the MCO must survive while maintaining confinement and subcriticality. Accelerations to be used for the design basis are listed in Table 3-1. The temperature range for these drops is 25 °C to 132 °C (77 °F to 270 °F) and the pressure range is 0.0 to 450 lb/in² gauge.
 - A 2-ft vertical drop of a loaded and sealed MCO onto flat reinforced concrete — The MCO lands on the bottom end and there is no credible possibility of a side-slap-down secondary impact of the MCO.

Table 3-1. Multi-Canister Overpack and Component Accelerations Resulting from Design Basis Accident Drops.

Component		"Piston drop" into cask (g) ^a	2-ft drop (g)	Drop into CSB tube (g)	Dropped with cask (g) ^b
Loaded MCO	Horizontal	N/A	N/A	N/A	101
	Vertical	25	54	Specification, 35, lower MCO Calculation, 28, upper MCO	27
	Corner	N/A	N/A	N/A	Lid down: 27.4 Lid up: 33.5
Mark IA basket within MCO	Horizontal	N/A	N/A	N/A	101
	Vertical	Bottom basket, 25 Other baskets, 25	34	Specification, 35	Bottom basket, 27 Other baskets, 27
	Corner	N/A	N/A	N/A	Lid down: Top basket, 27.4 Other baskets, 27.4 Lid up: Bottom basket, 33.5 Other baskets, 33.5
Mark IV basket within MCO	Horizontal	N/A	N/A	N/A	101
	Vertical	Bottom basket, 25 Other baskets, 25	54	Specification, 35	Bottom basket, 27 Other baskets, 27
	Corner	N/A	N/A	N/A	Lid down: Top basket, 27.4 Other baskets, 27.4 Lid up: Bottom basket, 33.5 Other baskets, 33.5

Note: Angle of impact for drop within cask is 10.5° off vertical.

^aGravitational acceleration (g) is computed assuming the MCO is slowed by piston-like cushioning effect from air being squeezed through the 0.25-in. diametral clearance between the cask and the MCO and by the cushioning effect of adding water to the bottom of the cask. (Alternative acceptance criteria that do not require the use of water have been used at the CSB. Refer to Section 3.1.3.2 of this report.)

^bEdwards, W. S., 1998, *Safety Analysis Report for Packaging (Onsite) Multi-Canister Overpack Cask*, HNF-SD-TP-SARP-017, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

CSB = Canister Storage Building.
MCO = multi-canister overpack.

- A drop (worst-case orientation) of the MCO inside the sealed transportation cask — For an end-drop scenario, a secondary side-slap-down impact shall be considered. The MCO is physically constrained by the cask walls and remains in the cask. Note that for all drops when the MCO is in the cask, the MCO does not need to maintain a leaktight seal. The MCO shall be able to retain all particulate 2 mm in size or greater after any deformation occurs.
- A vertical drop of the MCO into the transport cask. Drop heights are not to exceed 21.5 ft. "Piston effect" shall be included.
- Vertical drops of the MCO into a CSB storage tube with and without another MCO already within the tube. The tubes will contain impact absorbers as required to reduce impact acceleration on the MCO and its internals. Each MCO acceleration is limited to 35 g within the CSB tubes.

Service level D allowable stresses may be exceeded for horizontal and corner drops while the MCO is in the cask. However, the criticality control measures in Section 4.19.3 of the MCO Performance Specification (Goldmann 2000b) shall be maintained. Exceeding service level D allowables while in the cask is justified for the following reasons.

- When the MCO is within the cask, the confinement is supplied by the cask and the MCO can be considered "out of service" as related to an ASME coded pressure vessel.
- The most severe corner drops due to large offsets have been addressed. These drops are facility-specific to the CSB and further detail can be found in HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Annex A, "Canister Storage Building Final Safety Analysis Report" (HNF 1999).

Table 3-1 has been prepared to indicate the various acceleration forces associated with the accidents discussed in this section and with other events. (The acceleration forces are expressed in "g"'s, with 1 g being equivalent to the gravitational force of an object on the surface of the earth.) Table 3-1 lists the acceleration forces allowable for different payload MCOs, during different drop scenarios, for different drop heights. The acceleration values presented are a combination of design criteria and analytical results of the drops applied to the baskets unless otherwise stated. In some cases, categories are further broken down into horizontal, vertical, and corner drops for the MCO inside the cask and for the MCO without a cask. Presentation of the acceleration values in Table 3-1 is supplemented in some cases with 2-ft drop heights that quantify the total energy available for structural material deformation. The table presents two bounding situations, one is related to force and the other related to total energy available for material deformation.

The limitation on acceleration without regard to drop height bounds the high-energy drops in which the drop is attenuated with an impact absorber. The maximum acceleration level correlates with the maximum force on the individual components, and survival is not related to the

total energy of the drop. MCO survival is assured as long as the impact absorber does its task by absorbing the drop's excess forces (in the form of mechanical work and the crushing of the absorber). The impact absorbers provided by the CSB are designed to limit the applied force on the MCOs to less than 680,000 lbf. The 680,000 lbf roughly corresponds to a 35-g acceleration. An impact limiter is provided for the bottom of the CSB storage tube and in between the MCOs in the CSB storage tubes.

The actual proposed design of the intermediate impact absorber (between the MCOs in the CSB storage tubes) indicates that the nominal crush forces are between 552,000 lb and 567,000 lb with a new maximum at about 624,000 lbf. Assuming these values, there will be less force to contend with on the overhead drops (i.e., down from the 680,000 lbf design value).

The actual full-scale testing of the impact absorbers was completed in late August 1999. The test results showed the resultant limiting forces to be less than 680,000 lb. The resultant limiting force of less than 680,000 lb will be applied to the bottom of the striking MCO and to the shield plug or cover cap of the stricken (bottom) MCO. Using the impact absorbers, the assumptions also may include no energy being absorbed in the MCOs. Acceptance criteria for the MCO shield plug area after these drops are in accordance with Sections 3.1.3.2 and 3.1.3.3 (material acceptance criteria) and include the following.

- Leakage rate increases of less than 100%, based on the original allowable leakage rate of the MCO at the affected area, are acceptable.
- The MCOs are both recoverable after the event.

The acceleration values associated with the 2-ft drop exceed the 35-g criterion, but the resulting damage is minimal because of the limited total energy available. Applying the energy to the MCO, the reacting systems material arrangement, mechanical advantage, and material properties are sufficient to survive the event without significant permanent deformation of the components. In general, not enough energy is available to fully load the structures to the point of significant deformation before the event is over. The acceleration forces resulting from this drop were analytically determined to be 54 g in the baskets.

For all accelerations and design basis stresses, the SNF has been modeled with the properties of stainless steel except for the scrap baskets, which have been modeled as having hydrostatic properties when externally loaded. In cases in which one component is dropped onto another (e.g., an MCO onto an MCO in the storage tube). The eccentricity of the drop is negligible and does not require consideration.

The loads resulting from the accelerations listed in Table 3-1 that are applied to the confinement boundary and have acceptance criteria originating from Subsection NB (ASME 1998) are evaluated in Section 3.2.2. There is a high probability of localized denting of the MCO shell if it interacts with the positioning ring inside the cask as a result of a horizontal drop accident. The localized denting expected in the MCO when it is subjected to a side drop while contained in the shipping cask has not been predicted by a detailed analysis because while

the MCO is within the cask, confinement is supplied by the cask and the MCO does not have to provide coded service during this time. The MCO can be considered "out of service" as related to an ASME coded pressure vessel. However, because of the high elongation capability of the material and the limited amount of distortion (thickness of the ring), no breach of MCO confinement will occur. Because this is an inward denting, the circumference criterion for the criticality control contingency is not violated. For all other design load conditions noted in Table 3-1, the criteria of Subsection NB (ASME 1998) are met. The shell also has been evaluated to demonstrate that the circumference restriction for criticality control purposes is not violated for any load condition.

The accelerations in Table 3-1 that apply to the Mark IA baskets are evaluated in Section 3.2.3. In addition to the criteria of Subsection NG (ASME 1998), the criterion of maintaining geometry for criticality control in sequential loadings, first the vertical loading and then the horizontal and/or corner loading, was imposed. Both the criteria for Subsection NG (ASME 1998) and the criterion for criticality control geometry are met for all design base load conditions. The evaluations demonstrate that the center tube does not move radially more than 2 in. It also has been demonstrated that the baskets do not deflect sufficiently to become uncoupled and allow fuel particles bigger than 2 mm in diameter to enter into the center void, although particles smaller than 0.25 in. in diameter should be uncommon. The Mark IV baskets are not evaluated in the same detail as the Mark IA baskets because the baskets provide no criticality control positioning structure. However, refer to Section 3.2.3.3 for the stress analysis. Refer to Chapter 6.0 for the discussion regarding criticality control for Mark IA and Mark IV fuel.

3.1.3.1 Transportation-Related Drop Analyses: The acceleration data given in Table 3-1 were obtained by comparing results from three sources, as noted below. The data reflect analytical experience that has evolved over the course of the SNF Project. The MCO design was initially set using maximum accelerations of 100 g for the vessel pressure boundary and 35 g to 50 g for the internal baskets with the variance due to the different MCO fuel payloads. These values were selected based on predicted drop energies over the transport route for the loaded MCO containers. To evaluate these initial specification acceleration limits, calculations were conducted by the following groups:

- Design agent for the cask and transport package
- Transport and packaging organization responsible for the safety analysis report for packaging
- SNF Cask/Transportation subproject, for purposes of oversight.

As might be expected with intricate nonlinear analyses, the results vary based on assumptions used in the packaging and impact medium modeling. The analyses confirmed the bounding nature of the original specification values but also suggested that those original values were high for the possible impacts during transportation of the MCO (Marlow and Shiraga 1997). The impact prediction methodology concentrated on accepted impact prediction processes.

Design Agent for the Cask and Transport Package. The design agent for the transport package evaluated transport-dictated drops for the effects of a hypothetical drop of the cask onto a concrete slab:

- A 30.0-ft end drop
- A 30.0-ft side drop
- A 30.0-ft center-of-gravity-over-corner drop
- A 30.0-ft shallow angle impact (slap down).

The impact analysis was based on methodology of the Electric Power Research Institute (EPRI) NP-4830, *The Effects of Target Hardness on the Structural Design of Concrete Storage Pads for Spent-Fuel Casks* (EPRI 1986), and EPRI NP-7551, *Structural Design of Concrete Storage Pads for Spent-Fuel Casks* (EPRI 1991). The impact analysis considered the mass and geometry of the cask but assumed it to be rigid compared to the concrete slab. The concrete slab properties and the cask geometry were used to determine the pad hardness parameter. Scale model drop testing at Sandia National Laboratories and full-scale cask drop testing in England have been performed in an attempt to "benchmark" the EPRI methodology. The preliminary results of the tests (end drops) show excellent correlation with the predicted results.

The EPRI reports provide force (applied to the cask) versus deformation (of the target) curves for different magnitudes of target hardness. The target hardness is defined as a set of parameters times the area of the impact surface. This impact area usually depends on the deformation. The following procedure is used to determine the maximum deceleration of the cask and deformation of the target.

1. A small target deformation is assumed.
2. The geometry of the cask relative to the target is used to compute the impact area for the given deformation.
3. The target hardness is computed for this target area.
4. The data in the EPRI reports is used to determine the force associated with the deformation.
5. The energy absorbed in the increments in deformation is evaluated as the area under the force-versus-deformation curve for the increment in deformation.
6. The deformation is increased and steps 2 through 5 repeated.
7. This process is continued until the absorbed energy equals the weight times drop height. This is the final resulting solution for the force and deformation.

The following technical data define the cask, concrete slab, and soil:

Maximum weight of loaded cask = 60,000 lb

Concrete elastic modulus = 3.6×10^6 lb/in²

Ultimate concrete strength = 4,000 lb/in²

Poisson's ratio of concrete = 0.17

Concrete pad thickness = 8 in.

Rebar yield strength = 60,000 lb/in²

Subsoil modulus = 28,000 lb/in²

Poisson's ratio of soil = 0.49.

The concrete slab was assumed to contain 0.875-in. diameter rebar (#7) on 12-in. spacing top and bottom, two-way, with 2-in. coverage.

Transport and Packaging Organization Responsible for the Safety Analysis Report for Packaging. The Hanford Site transport packaging organization determined the drop accident impact accelerations, deformation, and stresses of the total package. Drop stresses and deformation of the cask were analyzed using the ABAQUS/Explicit Finite Element Analysis program³ (ABAQUS 1995a). Data for benchmarking of the analytical impact models were obtained from cask drop tests onto various concrete and soil surfaces, which were conducted by Sandia National Laboratory in 1986. Actual cask impact test data are summarized in a Sandia National Laboratory report (Gonzales 1987), and pertinent data are shown in Table 3-2.

Table 3-2. Sandia National Laboratory Target Hardness Test Results.

Test		Maximum recorded acceleration (g)	Penetration (in.)
Concrete runway	44 ft/s	480	0.25
	66 ft/s	900	4
	88 ft/s	1,000	8
Concrete highway	44 ft/s	350	4
	66 ft/s	--	4
	88 ft/s	7,500	19

Note: Information in this table is from Gonzales, A., 1987, *Target Effects on Package Response: An Experimental and Analytical Evaluation*, SAND86-2275, Sandia National Laboratory, Albuquerque, New Mexico.

³ABAQUS is a trademark of Hibbitt, Karlsson, and Sorensen, Incorporated.

The test data were used to develop and benchmark realistic two-dimensional and three-dimensional models simulating cask impacts onto concrete surfaces with the ABAQUS/Explicit finite element analysis program (ABAQUS 1995a). The resulting computer simulations were used to determine concrete and soil target behavior and simulation parameters.

Comparisons between three, two-dimensional, calibrated simulations and test data are shown in Table 3-3 to demonstrate the validity of the ABAQUS/Explicit finite element analysis modeling methods for cask impact evaluation (ABAQUS 1995a). The two-dimensional model used both the runway concrete and highway concrete cask-impact test data to study material behavior and develop simulation parameters for target models. In the two-dimensional model, the concrete is modeled as a uniform isotropic material and the soil as nonlinear spring elements.

Table 3-3. Two-Dimensional Model Comparison of Highway and Runway Concrete Impact.

Impact velocity (ft/s)	Target	Calculated peak load (g)	Measured peak load (g)	Calculated penetration (in.)	Measured penetration (in.)
44.0	Concrete highway	583	350	3.7	4.0
44.0)	Concrete runway	814	480	0.3	0.25
88.0)	Concrete runway	1,128	1,000	5.3	8.0

Results comparing three-dimensional, calibrated simulations with actual test data are shown in Table 3-4 to demonstrate the accuracy and realistic representation of the ABAQUS/Explicit finite element analysis modeling methods used in the evaluation of casks (ABAQUS 1995a). The three-dimensional model differs from the two-dimensional model in that the soil and concrete are represented by eight-node brick elements. These changes were made for a more accurate representation of the concrete and soil behavior. Comparison of the two-dimensional and three-dimensional modeling results with actual test data demonstrates the three-dimensional model substantially improves simulation accuracy of cask impacts onto concrete surfaces.

The ABAQUS/Explicit finite element analysis computer code (ABAQUS 1995a) evaluates impact by modeling two parts, a projectile and target. The ABAQUS/Explicit computer code is a dynamic finite element analysis program that can be used for elastic, plastic, and inelastic analysis of structural members. The primary feature of ABAQUS/Explicit is that it has the ability to solve impact problems. The projectile was defined as the cask body, closure lid, closure lid bolts, and contents (MCO and SNF). The target surface was defined as the reinforced concrete pad and soil. In each case, the contact force is calculated from these two impacting parts.

Table 3-4. Three-Dimensional Model Comparison of Runway Concrete Impact.

Impact velocity (ft/s)	Calculated peak load (g)	Measured peak load (g)	Calculated penetration (in.)	Measured penetration (in.)
66.0	894	900	4.7	4.0
88.0	1,343	1,000	8.7	8.0

The projectile is the loaded cask defined as basically a hollow right circular cylinder with an outer diameter of 40.57 in. and length of 170.25 in. The contents cavity is 25.19 in. in diameter and 160.5 in. in length. Nominal wall thickness of the cylindrical portion of the cask body is 7.31 in., which provides both structural integrity and shielding. At the top end of the cask body are 12, equally spaced, 1.5-in.-6 unified national course thread bolt holes, which mate to the hat-shaped closure lid. As an integral part of the cask, the body is fitted with a 2.25-in.-thick, 1-in.-tall shear ring on the top end radial surface. (See Figure 3-1 for a sketch of the cask-MCO package.)

The gross weight of the loaded cask without water is 58,430 lb, with the center of gravity of the package located 83.6 in. from the bottom. Empty weight of the cask is 38,960 lb, with the center of gravity located 81.82 in. from the bottom of the package. Weights of the individual components are shown in Table 3-5.

Table 3-5. Component Weights.

Component	Weight (lb)
Cask body shell	34,300
Cask body bottom	2,270
Closure lid	1,890
Lifting attachment	500
Dry weight of loaded multi-canister overpack	19,470
Gross weight of cask (without water)	58,430

The concrete-soil target is 8 in. thick with reinforcing steel bars. For the analysis, the reinforcing steel bars are smeared into the concrete elements with the ABAQUS *REBAR command (ABAQUS 1995a). Depth of the soil region is 100 in. The soil constitutive model is a Drucker-Prager plasticity model developed originally for analysis of underground tanks.

A package impact in the orientation expected to cause maximum damage envelops the effects of Hanford Site drop accident transport conditions. The scenarios are free drops from a height of 30 ft with a dry MCO and 21 ft with a water-filled MCO onto an 8-in.-thick reinforced concrete surface with concrete strength of 4,000 lb/in², soil modulus of 28,000 lb/in², and No. 7 rebar concrete reinforcement with a yield strength of 60,000 lb/in², spaced 12 in. apart with 2-in. cover.

The structural adequacy of the transport package was demonstrated by showing the stress in the confinement boundary did not exceed the limits specified for Level D service limits in the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Class 1 nuclear components (ASME 1998). The stresses due to accident conditions were determined using the ABAQUS/Explicit finite element analysis computer code (ABAQUS 1995a). Although peak accelerations were determined, they are not used in evaluation of the stresses. As determined from British Nuclear Fuels Limited impact tests on concrete (Stokley and Williamson 1996), the use of peak acceleration for quasi-static evaluations is overly conservative.

To illustrate the overall cask response during impact, the deceleration is determined by two methods. The most conservative method is to divide the total reaction force of impact by the total cask assembly weight and define the peak deceleration as a function of time. The other method is based on the change in velocity over the estimated time to rebound. This time-averaged deceleration over the impact period more accurately reflects the actual cask global behavior by accounting for the cask response time during the impact. These decelerations are provided only for quasi-static evaluations. Dynamic evaluations of the cask-MCO loadings, such as stress and strain determinations, are performed directly by the ABAQUS/Explicit (ABAQUS 1995a and 1995b) finite element analysis computer code.

As with an actual cask test, the simulated cask impacts result in noise from high-frequency strain waves, similar to those recorded during instrumented cask drop tests. The frequency magnitude of these strain waves is such that the structure is not capable of reacting to them. In recognition of this behavior, International Atomic Energy Agency Safety Series 37 (IAEA 1990) recommends filtering out frequencies higher than 100 Hz to 200 Hz for a heavy package drop test. In this analysis, the deceleration time histories are processed through a two-stage filtration process using a six-pole sine-Butterworth filter. The first filtration process filters out numerical induced noise by filtering frequencies above 1,259 Hz. The second filtration process uses a cutoff frequency recommended in Safety Series 37 (IAEA 1990). For the approximately 60,000-lb weight of this cask, the cutoff frequency is 150 to 300 Hz. For conservatism, all of the deceleration time histories are filtered with a cutoff frequency of 500 Hz. In some cases a sharp spike, resembling a high-frequency noise, appears at time zero. This is a result of interaction among the residual forces from the myriad number of contact surfaces. Consequently, these are computational anomalies that do not represent the actual cask response. Filtering of higher frequencies eliminates these numerical anomalies.

SNF Cask/Transportation Subproject. The SNF Project conducted oversight drop accident analyses to verify that the accelerations predicted by the organizations above were valid. The SNF Project also analyzed many other drop accident conditions dealing with the SNF Project

facilities. The analysis methodology was similar to that used by the packaging organization but was conducted independently and with different soil parameters. As stated earlier, later comparison of soil stiffness data to that used and benchmarked with Sandia National Laboratory test data showed the oversight analyses were conservative as the soil stiffness was high.

Based on results from acceleration analyses performed by the packaging organization using the ABAQUS computer program and the cask/transportation design agents using EPRI data, which showed similar acceleration values from impact, the earlier design basis acceleration values were changed to those shown in Table 3-1. The values in Table 3-1 are lower than those predicted by the project oversight analyses but are justified based on analyses showing the oversight calculations used an overly conservative soil stiffness under the concrete impact medium. The oversight calculations produced accelerations that were conservative, as confirmed by a study on the effects of soil constitutive properties on impact-induced accelerations (Marlow and Shiraga 1997). The soil effects were checked by the cask transportation organization in conjunction with the packaging organization. This resulted in the agreement that the packaging dynamic models with soil parameters were more closely related to actual soil effects. This agreement was confirmed by Marlow and Shiraga (1997) as follows.

"The soil constitutive model recommended by Dr. Marlow (Packaging) is sound and based upon yielding surface impact tests and studies from SANDIA (Gonzales, 1987), BNFL (Stokley, et al, 1996), and LLNL (LLNL, 1987). This recommended model is derived from material impact behavior studies conducted for calibrations of the ABAQUS/Explicit (HKSa, 1995) impact models developed for evaluation of the MCO cask. The models and representations of material impact behavior developed were verified from impact test data for small casks by SANDIA, large casks by BNFL, and yielding surface impact studies by LLNL."

3.1.3.2 Facility-Specific Drop Analyses. Accidents involving the mishandling of the loaded MCO or MCO and cask package have been analyzed for the complete handling cycle. The cycle consists of activities from cask lifts at the K Basins to placement of the MCO into the storage tubes at the CSB. These handling accidents can be basically divided into two categories: the handling of the cask in transport configuration (applies to all portions of the MCO life cycle before final removal from the shipping cask), and the handling of the MCO by the MCO handling machine (MHM) in the CSB. The approaches to analyzing the drop scenarios at the facilities have matured over the life of the SNF Project and have evolved with the MCO design.

In the early stages of the SNF Project, every effort was made to anticipate the normal operational and accident conditions that the MCO might encounter. Credible and appropriate scenarios were developed early to allow the design of the MCO to proceed and be evaluated. By predicting the drops that could occur, the MCO could be designed to be capable of withstanding quantified levels of stress. The most basic means of quantifying the drops was to identify the accelerations experienced by the MCO and its components as the result of the drops. Table 3-1 of this report was prepared to provide the allowable accelerations for different MCO payloads under different scenarios. The acceleration values presented in Table 3-1 are a combination of design criteria and analytical results. Section 3.1.3 of this report also discusses the acceptability

of exceeding the allowable values provided in Table 3-1. Rationale for accepting higher accelerations includes factors such as maintaining confinement, not violating criteria associated with the criticality control contingencies, and the fact that if the MCO is in the cask it can be considered to be "out of service" as related to an ASME-coded pressure vessel.

The following paragraphs summarize the initial approach used in evaluating the drop scenarios against the allowable accelerations and identification of the need for prevention or mitigation. The supplemental analyses approach and the modified acceptance criteria are than summarized to complete discussion of the drops.

Initial Drop Analyses. Various drop scenarios were analyzed by use of the ABAQUS computer program (ABAQUS 1995b), and the results were compared with Sandia National Laboratory test data to verify that the initially established MCO design values were met. The results of the drop analyses show that the initially established values were reasonable estimates and prudent design values. Potential drop situations that are facility-specific and under the control of those facilities were studied for the same comparisons with established design values. In all but three analyzed facility-specific cases, the established values bounded any previously predicted accelerations. Three facility-specific drops create accelerations in excess of the established design basis accident values. However, these drops all occur in the CSB and are associated with the offloading of the cask and/or the MCO from the cask. These conditions required either mitigation features or the use of alternative acceptance criteria. The following drop groups were subsequently found to exceed the acceleration values specified in the MCO Performance Specification (Goldmann 2000b).

- Group 1 — Cask-MCO drops onto the CSB receiving area floor

Analysis of the cask-MCO drops onto the CSB receiving area floor is documented in HNF-SD-SNF-DP-010, *Cask and MCO Drop onto the Transport Trailer Edge with Subsequent Horizontal Slap Down onto the CSB Receiving Area Floor* (Moore 1997a), HNF-SD-SNF-DP-007, *Multi-Canister Overpack/Cask Drop Analysis File Documentation*, Appendix G, "Cask and Multi-Canister Overpack Drop Simulation at Canister Storage Building Load Out Area" (Chenault 1997a), HNF-3158, *Detailed Simulation of Cask and Multi-Canister Overpack Drop onto the Transport Trailer Edge with Subsequent Horizontal Slap Down onto the Canister Storage Building Receiving Area Floor* (Moore 1998a), and HNF-3243, *Detailed Simulation of Cask and Multi-Canister Overpack Vertical Drop onto the Canister Storage Building Receiving Area Floor* (Moore 1998b).

- Group 2 — MCO drops (a) into the cask and (b) onto the cask chamfer

Analysis of MCO drops into the cask and onto the cask chamfer is documented in HNF-SD-SNF-DP-007, Appendix C, "Multi-Canister Overpack Drops into the Cask and onto the Cask Chamfer" (Chenault 1997a). The result of the concentric drop into the cask is quite severe.

- Group 3 — Cask-MCO drop into CSB transfer pit

Analysis of the cask-MCO drop into the CSB transfer pit is documented in HNF-SD-SNF-DP-007, Appendix O, "Cask Drop Simulation into Canister Storage Building Transfer Pit" (Chenault 1997a).

These three drop groups previously would have required some type of mitigation features to meet the then current acceptance criteria. One drop at the K Basins also was identified as exceeding the 35 g/100 g limit of the Performance Specification (Table 3-1 of this report). However, that drop occurred while the MCO was in the cask and therefore out of service relative to the service level D requirements of the ASME Code (ASME 1998). Nevertheless, the drop was analyzed (M&D 2000) to ensure that criticality was not an issue.

Group 1 drops are addressed in the "Supplemental Drop Analyses and Modified Acceptance Criteria" section.

The Group 2b drop listed above is now mitigated by revisions to the MHM that reduce the amount of eccentricity and place a funnel in the cask entrance to guide the MCO. The concentric drop into the cask remains unmitigated. Analysis has been conducted that shows the accelerations resulting from the unmitigated concentric MCO drop back into the cask (Group 2a) exceed the current specification allowable (Goldmann 2000b) but do not cause breach of the pressure confinement boundary. Since this condition satisfies the confinement requirement of ANSI/ANS-57.9-1992, *Design Criteria for Independent Spent Fuel Storage Installation (Dry Type)* (ANSI/ANS 1992), this drop condition is acceptable. ANSI/ANS-57.9-1992 allows a category IV drop such as this be acceptable when one confinement boundary beyond the MCO exists if an MCO is dropped and breached. The Group 3 drop is mitigated by an impact limiter placed in the bottom of the transfer pit. Refer to the facility safety analysis reports for more detail on these specific accidents and mitigation features.

Two independent sets of analyses show that for select vertical and side drops resulting in higher accelerations than allowed in Table 3-1, criticality control features will remain within allowable deformation limits. The conclusion drawn from analyses documented in RFSNW-9754340 (Marlow and Shiraga 1997) is that higher accelerations, up to 300 g for a vertically oriented MCO, do not result in a violation of criticality control deformation limitations. Additional analyses documented in HNF-1896, *MCO Basket Deformation and Damage During a Horizontal Slap Down onto the CSB Receiving Floor* (Moore 1997b), show that horizontally oriented MCOs, when dropped, do not cause criticality deformation limit violations up to 440 g. Analyses have been completed (Moore 1998a, 1998b) that show the MCO or cask will not lose the ability to maintain confinement.

The present MCO design has been demonstrated to meet ASME Code criteria (ASME 1998) for drops resulting in accelerations shown in Table 3-1. If a drop produces accelerations equal to or less than those shown, the MCO package can be handled as suitable for service and continue through the storage steps. If a drop were to exceed those accelerations, the SNF Project, before moving the MCO, would review the need for corrective actions to allow for

continued processing. All facilities having the potential for drops exceeding those addressed in the MCO Performance Specification (Goldmann 2000b) must have designed into the facility mitigating features that will hold the accelerations to the MCO Performance Specification acceleration limits or meet criticality requirements.⁴

Supplemental Drop Analyses and Modified Acceptance Criteria. Cask and MCO handling at the CSB presents special challenges. The floor over which the cask and MCO are handled is constructed of very high strength concrete and is 5 ft thick. This hard, strong surface means that drops at the CSB have the potential to impose on the cask and MCO conditions more extreme than drops at other points of the cycle. As a result, considerable attention has been given to these events and additional analyses have been performed. The lifting yoke used to unload the cask is designed to limit the lift height to 60 in. This in turn limits the maximum drop onto the floor to 60 in. A spectrum of drops including corner drops and slap downs has been analyzed, and the analyses have demonstrated that the cask survives and retains its confinement and pressure boundary functions for all drops of 60 in. or less. Preservation of these functions prevents release and retains a barrier to ingress of moderator into the cask-MCO system.

Longer drops can be postulated at the CSB pits. A safety-significant impact absorber mitigates drops into the service pit. Design features and administrative controls are employed to reduce the potential for drops into the other pits.

Handling of the MCO in the MHM includes lifting the MCO out of the cask, transporting the MCO through the CSB, and lowering the MCO into the weld station and the storage tube. Drop accidents can be postulated at each step in the handling cycle. Long drops are possible during cask unloading and at the sampling/weld station and storage tubes. At the sampling/weld station and the storage tubes, the safety-significant impact absorbers reduce deceleration to ensure the MCO is not damaged. For the drop back into the cask, analysis has shown that the confinement and pressure boundary functions of the MCO survive the drop. For an eccentric drop into the storage tube in which the MCO impacts the tube flange, analysis has shown that the MCO survives this impact. For eccentric drops at the cask unloading station and sampling/weld station, lead-ins on shield blocks have been designed to minimize damage to the MCO. The potential exists for one or more of these drops to exceed the allowable accelerations shown in Table 3-1. Therefore an alternative approach, including modified acceptance criteria, was needed for evaluating the CSB potential drop scenarios.

Drops that exceed the allowable accelerations may be evaluated using the failure criteria provided in the Expert Panel Report attachment to SNF-5204, *Analysis for Eccentric Multi-Canister Overpack Drops at the Canister Storage Building* (Tu and Hollenbeck 1999). The report states the failure criteria as follows:

⁴Criticality is controlled by MCO deformation limits as described in the MCO Performance Specification (Goldmann 2000b). When deformations are limited to less than the control limits, criticality control contingencies remain and thus criticality is not a concern.

“Failure Criteria, Conservative failure strain levels were predicted for both welds and parent 304L stainless steel material, accounting for scatter in properties and multi-axial stresses and strains, and using methods in Attachment 1. For a triaxiality factor of 1.0 (uniaxial tension), the above values are 21% for weld metal and 40% base metal. For a triaxiality factor of 2.0 (maximum value for biaxial tension at surface) the above values are 10% and 20%. For negative values of triaxiality factor (significant compressive stress) that are characteristic of the major portion of the strain accumulation in the drop, the values are 42% and 80%, again for weld metal and base metal respectively.”

The failure criteria statement above references Attachment 1, which is a paper titled “Prediction of Failure Strain,” and includes eight reference citations. The paper describes the rationale for varying failure strains under differing triaxial conditions using the Manjoine correlation. Both the Expert Panel Report and Attachment 1 are included in SNF-5204 (Tu and Hollenbeck 1999).

The acceptability of the predicted strain levels for the various drop events is based on the comparisons of the predicted multiaxial principal strains and the associated maximum plastic effective strain levels to conservative failure criterion. This failure criterion employs lower bound failure strains that account for the material parent and weld minimum tensile elongation (mean minus two standard deviations), the stress and strain triaxiality, the metal temperature, the strain rate, and reasonable margins.

The variability of ductility caused by the triaxiality is accounted for by using the Manjoine correlation. The computer analysis uses and reports “true” strain values, so further reductions for this, as well as adjustments for temperature, strain rate, and triaxiality are applied to the above conservative failure strain levels.

The predicted maximum strain levels are examined for their local nature and for the tensile or compressive nature of associated principal stresses and strains. Through-wall cracking is less likely with local and compressive states of stress and strain.

Generally, margins of 20% or more between predicted strain levels and their associated local tension cracking failure strain levels are considered acceptable and reasonable. In the application of the above failure criteria to a beyond design basis accident, the accident’s probability of occurrence and its radioactive release consequence may be considered. After establishment of the risk profile, alternate criteria may be used after development of appropriate technical justification and acceptance by the owner.

The potential drops that may occur at the CSB have been carefully evaluated and in many cases additional analysis performed and modified criteria applied. Because these drops are of a very facility-specific nature, no attempt is made in this report to provide details of the analyses. Furthermore, facility-specific analyses of drops at the CSB have demonstrated that the frequency of a drop is in the unlikely frequency range. The calculated consequences for an unlikely drop at

the CSB satisfy the evaluation guidelines. Additional information is contained in Chapter A3.0 of HNF-3553, Annex A (HNF 1999).

3.1.3.3 Bounding Drop Accident. A review of the cask-MCO lift and transfer paths was completed for the K Basins and for the CSB (FDH 1999). This review identifies the maximum lift heights and the impact media located at the lift locations. Analysis (Chenault 1999) conducted using the information above initially showed that the bounding drop (i.e., the drop producing the most severe acceleration of the package for all SNF cask-MCO lifts) occurs at the CSB. This drop location was initially found to be the transfer path from the cask trailer to the receiving pit, and the drop was a 40-in. drop of the MCO in the cask onto 5-ft-thick reinforced concrete that had a compressive strength of 9,000 lb/in². The resulting decelerations were a vertical end-on drop of 210 g and a horizontal slap down drop of 440 g. Subsequent analysis has analyzed this same drop location from a height of 60 in. and demonstrated that cask confinement is maintained. In addition, a drop of the MCO back into the cask at the CSB has shown even higher accelerations than evidenced in the 60-in. drop. However, this drop into the cask also showed that the MCO maintained confinement using the failure criteria discussed above.

This drop of the MCO into the cask (Group 2) with no mitigation results in accelerations greater than any credible drop onto the reinforced concrete at CSB and is therefore now considered to be the bounding drop at CSB. The accelerations resulting from this drop also exceed those calculated for any drop at the K Basins. However, because of the different acceptance criteria used in the CSB and K Basins, a bounding drop for the K Basins is also identified. That drop is the drop of the MCO while in the cask at the K Basins. Following the results of Marlow and Shiraga (1997), Moore (1997b, 1998a, 1998b), and M&D (2000), this bounding K Basins drop does not violate criticality control features; confinement will be maintained; and the drop is therefore acceptable. The drop will not, nor does it have to, meet the ASME Code allowables (ASME 1998) because the coded pressure vessel (the MCO) is out of code service while enclosed within the transportation cask. No drops are considered to be credible for the CVDF because the cask and MCO do not leave the transport trailer.

3.1.4 Multi-Canister Overpack Weights

Five baskets loaded with Mark IV fuel assemblies, which are approximately 26.1 in. long, or six baskets loaded with Mark IA fuel assemblies, which are approximately 20.9 in. long, can be stacked within the MCO. Fuel fragments greater than 0.25 in. in cross section are loaded into the MCO using scrap baskets. HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for Spent Nuclear Fuel Processing and Storage Facilities* (Kessler 2000), allows up to two scrap baskets, one at each end of the MCO. The limits on scrap basket loading are defined in HNF-SD-SNF-CSER-010, *Criticality Safety Evaluation Report for the K Basin Fuel Retrieval Subproject* (Kessler and Peck 2000). In addition, no scrap basket can be loaded to a weight exceeding that of a maximum weight fuel basket.

Detailed weight calculations for each component of the MCO are provided in HNF-SD-SNF-DR-003, *Multi-Canister Overpack Design Report* (Goldmann 2000a). Table 3-6 presents a summary of the nominal and maximum weight calculations for the MCO under different load conditions. The MCO center of gravity for various configurations is shown in Table 3-7. The measurements are in inches from the bottom of the MCO.

3.1.5 Multi-Canister Overpack Dimensions

The maximum height of the MCO with the cover cap absent is nominally 160 in. and 166.3 in. with the cover cap in place. In no case, including post-accident conditions, is the MCO inside circumference below the bottom of the shield plug allowed to exceed 73.04 in. (23.25 in. π) (Goldmann 2000b). The MCO shell is allowed a 25.31 in. maximum as-built outer diameter above the 148 in. elevation measured from the MCO bottom. The MCO Design Report (Goldmann 2000a, Appendix 5 and Appendix 11) and Section 3.2.2 of this report show that the inside circumference below the bottom of the shield plug is maintained at less than 73.04 in., even following the DBA events defined in the MCO Performance Specification (Goldmann 2000b). The only event that has the potential for any significant distortion of the circumference of the shell is the side drop in the cask, and the distortion in that case is localized in the area of the collar in the cask. The distortion would be inward (localized denting) and hence the circumference requirement would not be violated.

3.1.6 Multi-Canister Overpack Shell and Closure Design

The MCO shell assembly consists of a bottom baseplate with a low point in the center for facilitating the removal of water, a cylindrical 0.5-in.-thick shell, and a forged collar at the top that mates with the shield plug and locking ring. The forged collar permits full access to the inside cavity for stacking the fuel baskets. At the bottom of the cavity, six pie-shaped basket support plates support the bottom basket and maintain a 1.24-in. nominal distance between the basket and the bottom surface. The empty shell can be loaded into the transportation cask by installing the locking and lifting ring in place and grappling the lift ring on the locking ring in a manner similar to that used in handling the loaded MCOs. The MCO may be turned upright with standard rigging (e.g., slings) although a turning fixture may facilitate the operation. The MCO should not be handled or lifted without the lifting ring in place. The use of other fixtures to lift the MCO increases the potential for damaging and misshaping the collar area so that insertion of the locking ring may be difficult.

The MCO shield plug assembly rests on a seal ledge on the inside of the MCO collar. The shield plug has a groove in the ledge mating surface that holds the main seal and prevents excessive deformation of the seal. The shield plug is held in place by the locking ring assembly, which threads into the collar with a double lead buttress thread. Once the locking ring is in place, eighteen 1.5-in.-diameter set screws are torqued, generating a force between the shield plug and the locking ring. The force seats the metal main seal and maintains a minimum sealing load on the

Table 3-6. Multi-Canister Overpack Weight Summary for Various Components and Conditions. (2 sheets)

MCO component	Weight (lb)	Maximum weight (lb)
Shell	1,472	1,517
Collar	197	205
Bottom plate	256	260
Support plate	10.4	11.2
Guide cone	3.5	3.6
Shield plug	781	788
Locking ring and set screws	403	417
Basket stabilizer ext	25	26
Internal filters	17	50
Guard ring	80	88
Guard plate	159	164
Process tube	45	47
Port cover plates	27	29
Valve port 2	1.7	1.78
Valve port 3	1.7	1.78
Valve and rupture disk assembly	1.55	1.63
Canister cover (cap)	509	521
Mark IA fuel basket/fuel ^a	472/1,906.08	NA
Mark IA scrap basket	625	NA
Mark IV fuel basket/fuel	228/2,990.52	NA
Mark IV scrap basket	432	NA
Mark IA water	891	+138 without plug
Mark IV water	902	+137 without plug
Empty Mark IA/Mark IV no fuel, condition 2	3,478	3,608
Mark IA condition 1, loaded and H ₂ O	17,237	17,296
Mark IV condition 1, loaded and H ₂ O	19,071	19,129
Mark IA/Mark IV condition 1, empty	1,940	1,999
Mark IA condition 2, loaded and H ₂ O	18,637	+130

Table 3-6. Multi-Canister Overpack Weight Summary for Various Components and Conditions. (2 sheets)

MCO component	Weight (lb)	Maximum weight (lb)
Mark IA condition 2, loaded and dry	17,746	17,876
Mark IV condition 2, loaded and H ₂ O	20,471	20,602
Mark IV condition 2, loaded and dry	19,569	19,700
Mark IA condition 3, loaded and H ₂ O	18,206	18,320
Mark IV condition 3, loaded and H ₂ O	20,040	20,154
Mark IV condition 2, dry with cap	20,079	20,220
Mark IA condition 2, dry with cap	18,255	18,397

Note: Columns contain nominal values with a maximum obtainable with the variation. The values in Appendix 3 of Goldmann, L. H., 2000a, *Multi-Canister Overpack Design Report*, HNF-SD-SNF-DR-003, Rev. 3, Fluor Hanford, Incorporated, Richland, Washington, are slightly less than shown because of valve and support plate weight differences.

Condition 1: All components that can hold water. This excludes the shield plug plus attachments, locking ring and screws, and process tube.

Condition 2: All components of a working MCO except the cover cap.

Condition 3: All components except the locking ring and screws, cover plates, and cover cap.

*The Mark IA MCO holds six baskets.

*The Mark IV MCO holds five baskets.

MCO = multi-canister overpack.

Table 3-7. Loaded, Dry Multi-Canister Overpack Center of Gravity.

MCO configuration	Center of gravity without cover cap	Center of gravity with cover cap
MCO loaded with Mark IA fuel	79 ± 1	81.5 ± 1
MCO loaded with Mark IV fuel	79 ± 1	81.2 ± 1
Empty MCO	106.1	113.4
MCO cover cap	NA	163.5

The center of gravity of the baskets is assumed at one-half of the basket height.

seal during all loading conditions. The evaluation of the set screws and their required torquing is discussed in Section 3.2.4. Sealing details can be found in Appendix 13 of the MCO Design Report (Goldmann 2000a).

The MCO shall relieve internal pressure in excess of 150 lb/in² gauge while it is flooded with water. A safety-class rupture disk embedded in the shield plug provides overpressurization protection with a minimum 1-in.-diameter flow area to accommodate pressure relief. The rupture disk is covered with a removable 1-in. minimum thickness orifice plate to protect the disk from potential overhead strikes (e.g., from dropped tools, gauges, equipment). Once bulk water is removed from the MCO and the cold vacuum drying process is completed, the rupture disk orifice plate is replaced with a blind cover plate, and the disk becomes inactive and remains inactive during cover cap welding and interim storage at the CSB.

The MCO is designed to incorporate a final welded cover cap over the shield plug. The cover cap is a 304L stainless steel forging that mates with the MCO collar. The cap is welded to the MCO shell assembly, and the weld is designed and performed in accordance with ASME Boiler and Pressure Vessel Code criteria (ASME 1998). Weld examination will include a liquid penetrant inspection of the root and final passes as a minimum. The lifting ring on the cover cap is the same diameter and thickness as the lifting ring on the locking ring. The MCO is capable of being lifted by the lifting ring on the cap when engaged by the grapple on the MHM (Goldmann 1998). The cap is capable of withstanding the pressure rating of 450 lb/in² gauge at 132 °C (270 °F) and meeting the drop criteria for the drop into the CSB tubes in accordance with Table 3-1. The cover cap is fitted with a recessed threaded plug for helium leakage rate test and other purposes. Discussion of the weld and of the lifting ring that is attached to the cover cap is found in Section 3.2.4.

3.1.7 Multi-Canister Overpack Fuel Basket Design

The MCO fuel baskets are categorized into two major configurations: baskets for intact fuel element sections and baskets for scrap fuel (fragments). Baskets must maintain criticality control for the more highly enriched Mark IA fuel. These basic requirements lead to four different basket types.

- Type 1 holds 48 intact Mark IA (higher enrichment) fuel elements and has a criticality control contingency exclusion void (center post) built into the basket (see Section 4.19.3 of HNF-S-0426 [Goldmann 2000b]).
- Type 2 holds 54 intact Mark IV fuel elements and does not need the exclusion void.
- Type 3 holds Mark IA (higher enrichment) scrap fuel (fragments) and has a criticality control contingency exclusion void built into the basket (see Section 4.19.3 of HNF-S-0426 [Goldmann 2000b]).
- Type 4 holds Mark IV scrap fuel (fragments) and does not need the exclusion void.

Single-pass reactor (SPR) fuel will be loaded into a Mark IA criticality control basket that has been fitted with an SPR fuel rack instead of the Mark IA fuel rack.

Loaded baskets can be easily and safely handled in the basin water. Lead-ins and alignment mechanisms allow the baskets to be loaded and nested in the MCO in the K Basins. All baskets have a central support tube for axial support during lifting and for process tube protection. The baskets stack inside the MCO with the basket centerlines coincident with the MCO centerline. While stacked inside the MCO, the baskets allow for insertion of the long axial process tube down the MCO centerline. The long process tube is used for water draining and gas transport, as needed. The shield plug assembly has an axial stabilizer that engages the top basket and allows for differential thermal expansion conditions. The baskets are compatible with the MCO materials relied on for confinement and with the fuel during the expected temperatures, pressures, and atmospheres inside the MCO during handling, shipping, drying, and storage.

The Mark IV and Mark IA storage and scrap baskets are constructed of 304L stainless steel and aluminum. Other scrap basket materials include copper for thermal conductivity as described in Section 4.12.6 of the MCO Performance Specification (Goldmann 2000b). All baskets are annular open-top containers with a maximum outer diameter of 22.625 in. at 25 °C (77 °F). All baskets support the fuel at 1.0 g while at 132 °C (270 °F). The basket grapple interface for all baskets is a 0.13-in. deep by 1-in. long radial groove beginning 1.84 in. from the top end of the basket center tube. Basket sizing accommodates a 0.3 in. nominal clearance between the top of the fuel elements and the bottom plate of the basket above.

Mark IA fuel has a higher U_{235} enrichment than Mark IV fuel. The structural integrity of the Mark IA basket is required for criticality control contingency whereas structural integrity for criticality purposes is not required for the Mark IV fuel basket. Therefore, the Mark IA fuel storage and scrap baskets have been designed to the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME 1998), under the component safety group as guided by NUREG/CR 3854 (NRC 1984). The designs meet service level A requirements for normal operating loads and service level D requirements for accident conditions under the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME 1998). During accident conditions the baskets designed for Mark IA fuel and Mark IA fuel fragments and scrap maintain the criticality control contingency features defined in Section 4.19.3 of the MCO Performance Specification (Goldmann 2000b).

Evaluation of the baskets for the load conditions defined in the MCO Performance Specification (Goldmann 2000b) is found in Section 3.2.3. The combined requirement of sequential loading from the vertical and horizontal drop events is addressed by conforming to both the ASME Boiler and Pressure Vessel Code service level D requirements (ASME 1998) and the MCO Performance Specification criticality requirements (Goldmann 2000b). This is necessary to ensure that the basket center tubes do not move more than 2 in. radially and that basket axial deformations are small enough to prevent the baskets from becoming disconnected from one another, an event that possibly would permit fine fuel particles to enter the basket center tube exclusion void.

For consistency with the Mark IA structural evaluations, the Mark IV baskets have been evaluated to ASME Code, Section III, Subsection NG, criteria for the service level A loadings. No service level D loadings have been considered for the Mark IV baskets.

For the handling of both loaded and unloaded Mark IA and Mark IV baskets, the design meets the safety factors required by the MCO Performance Specification (Goldmann 2000b). Design and qualification of the basket grapple interface was performed by testing conducted by the DE&S Hanford, Incorporated, Cask/Transportation subproject, and the test results verified the adequacy of the MCO grapple and interface design. The testing was conducted to the 1995 requirements of ANSI N14.6. The 1995 ANSI N14.6 safety factors apply from 5 °C through 100 °C (41 °F through 212 °F). Both the Mark IA and the Mark IV baskets have been evaluated for lifting in accordance with equivalent criteria as outlined in the MCO Performance Specification (Goldmann 2000b).

3.1.8 Multi-Canister Overpack Shield Plug Design

The shield plug is a multi-functional component of the MCO. It provides a mechanical confinement boundary until the cover cap is welded in place. It also provides for axial shielding to allow personnel access to the top of the MCO for securing the mechanical closure as well as for performing the drying and process functions. The shield plug also retains the main seal, which seals against the MCO collar. The MCO shield plug assembly is designed to mate with the open end of the MCO shell, with the end effector on the top SNF fuel basket, and the cone (via the long process tube) at the MCO bottom. The shield plug assembly features an integrally machined, axisymmetric lifting ring with a 12-ton lifting capacity when gripped with six equally spaced grippers (1.97-in. tangential length by 0.66-in. radial contact length). The lifting ring will facilitate handling of the MCO package when it is unloaded from the transport cask, CSB storage tubes, and CSB weld stations with the MHM. Demonstration of the 12-ton capacity is discussed in Section 3.2.4 and in Appendix 6 of the MCO Design Report (Goldmann 2000a). The MCO lifting rim area of the lifting ring design meets the criterion described by the MCO Performance Specification (Goldmann 2000b). This criterion requires that any handling or lift features shall be capable of demonstrating a safety factor of three on material yield and five on material ultimate strength.

The shield plug assembly, which consists of the shield plug, guard plate, internal filters, process valves, and rupture disk, has a basket stabilizer extension that centers the top basket. The assembly has a minimum of 0.65 in. clearance above the fuel or fuel fragments at 22 °C (72 °F). The shield plug is held in place with a locking ring that has the aforementioned integrally machined, 12-ton capacity lifting ring.

The rupture disk was sized to allow a two-phase flow resulting from a worst case "runaway" corrosion event. An extensive analysis (Thurgood and Duncan 1997) was performed to ensure the 1-in. diameter relief passage and features are adequate. The cylindrical screens, which are under the shield plug and in the guard plate, have approximately 6 in² of open area. These screens are the beginning of the relief path from the MCO and have approximately eight

times the flow area of the 1-in. diameter passageway. The screens have 2-mm openings that, if fouled, would require particles greater than 2 mm to foul the screen. In a corrosion runaway event, the flow below the shield plug is not of sufficient velocity to sweep up heavy metal particles or solid corrosion products of significant size to blind the screen. But, if some particles were swept up to the screen, the finer materials would go through the screen and exit the MCO and the coarser materials would reside on and foul the screen. Without giving any credit to the flow through a potential built-up particle bed itself, the screen would have to be approximately 85% fouled to begin to impair the needed flow capacity. The rupture disk is capable of actuation with a fouled screen.

Internal process high-efficiency particulate air (HEPA) filters installed between the shield plug bottom and the guard plate support the vacuum drying outflows from the MCO. These filters meet the requirements of HNF-S-0556, *Specification for MCO Internal Filter* (McCracken 1997). The MCO Design Report (Goldmann 2000a, Appendix 10) verifies that the structural attachment will support a filter assembly weighing up to 50 lb for the maximum loading given in Table 3-1. The internal process filters and passage to the rupture disk are protected by the guard plate, which is capable of withstanding the drop accelerations in Table 3-1 (Goldmann 2000a, Appendix 1 and Appendix 5). The guard plate provides an air pocket that keeps the filters dry when the shield plug is inserted into the MCO under water and protects the filters from damage in a top down drop. The internal HEPA filters are put into the MCO for process reasons. The filter assembly requirements include flow rates, filtering efficiency, size, corrosion resistance, and other appropriate needs and features. Testing has been performed on these filters during prototype testing work for the CVDF and found adequate.

A full description of the filter specification is provided in HNF-S-0556 (McCracken 1997). The minimum filtering efficiency is set at 99.97% for a particle size of 0.3×10^{-6} m. The minimum flow is set at 100 standard ft³/min at 11.5 in. of water column pressure loss. The entire filter assembly is made of 316L stainless steel, which is very corrosion resistant under the expected environmental conditions of the MCO.

The process valves and rupture disk, located on top of the shield plug, are protected by port covers. The port covers have been assessed for the impact of a cask lid dropped in the worst orientation to show that the covers will protect the port-installed appliances. This analysis is discussed in Section 3.3.2 and in MECH007.97, *Simulation of SNF Transportation Cask Lid Drop onto MCO Process Port Cover Plate* (Abatt 1997).

3.1.9 Cover Cap

The MCO cover cap is an SA-182 304L stainless steel forging that is placed on top of the MCO after the completion of cold vacuum drying. The purpose of the cover cap is to provide a welded closure that is capable of meeting the confinement leakage rate criteria. The closure weld is a full-penetration weld located just below the top of the locking ring on the collar of the MCO shell. This weld is a field weld that cannot be radiographically inspected because of the configuration and contents of the MCO. The exterior of the cover cap is machined to have the

same lifting ring configuration as the locking ring so the containers can be handled after installation of the cover cap. The cover cap has one mechanically sealed penetration that will be aligned over port 2 on the shield plug to permit operation of that plug port after cover cap installation. The penetration is a plug with a mechanical seal that will allow venting of the cover cap, sampling of the atmosphere within the cover cap, as well as access for operation of the shield plug ports. The penetration is finally covered with a plate and welded to the cover cap.

3.1.10 Other Design Features

3.1.10.1 Materials. The shield plug assembly is fabricated out of 304L stainless steel, and the locking ring is made of 304N stainless steel to ensure compatibility. The MCO shell and collar are fabricated out of 304/304L dual-certified material to provide added strength and weldability. The process plugs are fabricated out of Nitronic 60 stainless steel. No ferritic materials are used in the design. All materials are specified as either ASME (SA) or ASTM (A) materials. Metal-to-metal galling is minimized by allowing the use of lubricants on threaded surfaces and on other closely fitting surfaces where operationally they would be permitted, such as between the radial surfaces of the locking ring and the shield plug collar. Harder materials such as Nitronic 60 are used for the threaded fasteners.

3.1.10.2 Welded Joints. All MCO pressure boundary welds are designed and produced to the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB (ASME 1998), except the field closure weld, which is a full-thickness weld that cannot be 100% radiographically examined. It is designed for 100% ultrasonic examination. This weld provides closure to a stabilized system, and the quality is ensured by welding qualification, by liquid penetrant inspection, and by leakage rate testing the attachment weld. The margins of safety for the weld are provided in Section 3.2.4.2 and the MCO Design Report (Goldmann 2000a, Appendices 5 and 11).

All MCO pressure boundary welds by the fabricator are in accordance with ASME Boiler and Pressure Vessel Code, Section III, Division 1, NB-3350 (ASME 1998). All welds are sufficiently smooth to enable easy decontamination. Butt welds are ground flush to within 0.03 in. of base metal. Weld joint designs avoid potential contamination traps to the greatest extent practicable. All MCO pressure boundary welds and welds bearing the weight of the fully loaded MCO will be nondestructively examined in accordance with ASME requirements.

The field weld joining the cover cap to the MCO shell is designed to permit a 100% ultrasonic examination. Flat surfaces behind the weld, a minimum of 1.10 in. below and 1.325 in. above the weld centerline, are required to facilitate this examination. In addition, a 30° half angle is required on the weld preparations for a 60° weld.

ASME code case N-595 will be used to certify the closure weld. The N-595 code case has been through the ASME committee approval process and has passed. The code case describes the conditions for a closure of a canister such as the MCO. This code case is appropriate to the MCO. The MCO's NRC equivalency document, HNF-SD-SNF-DB-005, *Spent Nuclear Fuel*

Project Multi-Canister Overpack, Additional NRC Requirements (Garvin 1998) allows use of this code case under certain situations, if the MCO project so chooses.

The N-595 code case determines the required dye penetrant and leakage rate testing of the welds as applied to the MCO. The code case's examination requirements will be strictly followed. The welds are de-rated according to the examination requirements path chosen by the project.

Early in 1998, the capability to examine the cover cap weld by ultrasonic testing was put into the design. Hence, the capability to examine the completed weld is in the design of the MCO. However, with the application of Code Case N-595, this weld will be dye penetrant examined and leakage rate tested in accordance with the N-595 case. Should a need be identified in the future to volumetrically examine this weld, the weld configuration can accommodate ultrasonic examination.

The cover cap attachment welds are pressure-retaining joints that eventually will bear the weight of the MCO. The cover cap attachment weld for each MCO is examined in accordance with the fabrication drawings and the fabrication specifications for the field weld. The MCO Performance Specification requires the cover cap attachment weld to be "examinable" by ultrasonic testing (Goldmann 2000b). However, the ultrasonic inspection of the joint is not a requirement and is not planned. The cover cap attachment weld is a field weld, which is performed at the CSB. The radiographic test requirement is applied to the fabrication welds of the MCO performed in the fabricator's shop.

The cover cap weld is performed in accordance with ASME code case N-595. The root pass is dye-penetrant tested, and each subsequent, approximately 0.25 in., lift is dye-penetrant tested. After the last welding pass is laid down, the weld is leakage-rate tested in accordance with ANSI 14.5-1987, *For Radioactive Materials — Leakage Tests on Packages for Shipment* (ANSI 1987). Helium is put into the newly created cavity under the cover cap through the test plug penetration. After the cavity is filled with helium, the test plug, with seal, is installed and tightened. Then the 25.31-in.-diameter girth weld is leakage-rate tested. The leakage-rate acceptance requirement from ASME code case N-595 is 1×10^{-4} standard $\text{cm}^3/\text{s-atm}$. However, the SNF Project has set the leakage rate acceptance requirement for this weld at 1×10^{-7} standard $\text{cm}^3/\text{s-atm}$. The last exposed lifts of the girth weld are dye-penetrant tested. The penetration test plug is leakage rate tested with the acceptance requirement also set at 1×10^{-7} standard $\text{cm}^3/\text{s-atm}$. The order of the previous two tests is not important. The cover plate for the test plug's penetration is installed and welded, with dye-penetrant testing performed after the root pass and subsequent, approximately 0.25 in., lifts.

3.1.10.3 Safety Requirements and Classification. The various components of the MCO are classified into safety classes in agreement with Section 3.1.2 and HNF-PRO-704, *Hazard and Accident Analysis Process*, as shown in Table 3-8. In addition, code requirements for pressure boundaries maintaining confinement are also considered. MCO safety structure and component designations are derived from the results of the analysis of the DBAs (Chapter 11.0) and criticality contingency (Chapter 6.0).

Table 3-8. Safety Functions and Failure Consequences of Multi-Canister Overpack Structures, Systems, and Components. (2 sheets)

System or component	Function	Safety designation	Failure consequences
Shell assembly, including collar, shell, and bottom	Criticality control, contain and protect SNF, protect MCO pressure boundary	SC	Release of radioactive contents that could exceed offsite exposure limits; loss of contingency protection against nuclear criticality accident
Shield plug	Criticality control, contain SNF, protect personnel	SC	Release of radioactive contents that could exceed offsite exposure limits
Rupture disk	Criticality control, protect MCO pressure boundary	SC	Damage or expansion of MCO boundary
Locking and lifting ring and bolts	Maintain pressure on main seal, allow for lift of loaded MCO	SC	Release of radioactive contents that could exceed offsite exposure limits
Cover cap	Seal MCO	SC	Release of radioactive contents that could exceed offsite exposure limits
Mark IA baskets	Maintain Mark IA SNF elements and scrap in a safe configuration (criticality control)	SC	Loss of double-contingency protection against nuclear criticality accident
Mark IV baskets	Hold Mark IV SNF elements and scrap	GS	No release consequences
Mark IA and IV scrap basket copper shroud, copper subassembly	Hold scrap and dissipate internally generated heat within the basket	SS	No release consequences; copper is not a structural component and is used for thermal conductivity
Plug valves	Accommodate gas flows through process ports in support of MCO processing	SS	Inability to process the MCO; release of radioactive materials into the environment that exceeds exposure limits
Internal process filter	Maintain most radioactive solid materials within the MCO	SS	Release of radioactive materials from the MCO; pressure buildup within the MCO; loss of defense-in-depth protection for release of radioactive materials
Long axial process tube	Allow bulk water removal, introduction of gases during processing, and reflooding, if necessary	SS	Inability to remove water from MCO; inability to introduce gases to process MCO; inability to perform processing that puts the MCO into a safe configuration
Screened process exit port	Connect to rupture disk as vent path, serve as backup process exit	SS	Failure of rupture disk to relieve internal MCO pressure
2-mm process tube and guard plate screens	Keep particles >2 mm in diameter in the MCO	GS	Particles larger than 2 mm may leave the MCO

Table 3-8. Safety Functions and Failure Consequences of Multi-Canister Overpack Structures, Systems, and Components. (2 sheets)

System or component	Function	Safety designation	Failure consequences
Seal	Seal MCO shield plug to shell	SS	Release of radioactive materials from the MCO; pressure buildup within the MCO; loss of defense-in-depth protection for release of radioactive materials
0.25-in. orifice cover plates	Regulate gas flow from the MCO	SS	Valve with rupture disk becomes nonoperational, causing possible release
Cover or orifice plate bolts	Maintain seal pressure	SC	Loss of double contingency protection against leakage
Cover plates	Provide leaktight seal, protect appliances	SC	Loss of double contingency protection against leakage
Guard plate on shield plug	Protect internal MCO process filter, and screens, provide basket centering	SC	Potential damage to filter, short process tube, and screen; provide center to top basket
Port valve/plug and cover plate seals	To seal appliances to shield plug	GS	Loss of pressure; potential release

GS = general service.

MCO = multi-canister overpack.

SC = safety class.

SNF = spent nuclear fuel.

SS = safety significant.

- Safety class
 - Criticality (Section 6.1) — MCO baskets, including features required for criticality control and structural integrity
 - Thermal runaway reaction (Section 11.3.3) — MCO pressure boundaries including penetrations, seals, and plugs
- Safety significant
 - Hydrogen explosion within an MCO (Section 11.3.2) — MCO pressure boundaries including penetrations, seals, and plugs
 - MCO internal pressurization (Section 11.3.3) — MCO pressure boundaries including penetrations, seals, and plugs.

The confinement boundary components are designed and fabricated to the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB (ASME 1998), and the Mark IA baskets are designed and fabricated to Subsection NG.

The safety designations "SC," "SS," and "GS" in Table 3-8 are equivalent to NRC classification categories A, B, and C, respectively. Quality assurance programs appropriate to the designations in Table 3-8 are implemented in the fabrication specifications for the MCOs and the baskets.

3.1.10.4 Nuclear Criticality Safety. The MCO design is to achieve and maintain a critically safe array throughout the MCO design life. A criticality safety value of 0.95 for k_{eff} is used for the MCO design, functions, and related activities (Garvin 1998). According to criticality analyses (Kessler 2000), this will be satisfied for MCOs containing Mark IA fuel by ensuring a nominal 6.625-in.-diameter steel insert (center post) at the longitudinal centerline of the MCO. This insert is a stainless steel bar with a 6.625-in. outer and 1.75-in. inner diameter. It precludes intrusion of fuel into the central region of the basket. The insert centerline does not deviate more than 2 in. from the MCO centerline. The MCO maintains these conditions during and after being subjected to drop accidents. MCOs containing Mark IV fuel do not require this insert. In addition, the MCO (for all fuel types) is capable of withstanding the effects of drop accidents with the maximum inside circumference not exceeding the limits allowed in Chapter 6.0 of this report and originating in Section 14.9.5 of HNF-S-0426 (Goldmann 2000b).

As stated above, the MCO and its contents have been evaluated for all loadings specified in Table 3-1. The evaluation shows that the k_{eff} limit is met.

3.1.10.5 Summary of Compliance with Requirements. The design of the MCO and fuel baskets is in full compliance with the requirements of the MCO Performance Specification (Goldmann 2000b). The compliance is demonstrated in the design drawings and in the evaluation of the design to the specified requirements as summarized in the following sections of this chapter

and as detailed in the MCO Design Report (Goldmann 2000a, Appendixes 2 through 17). Table 3-9 provides a list of the appendixes in the MCO Design Report and the correlated sections in this chapter.

Table 3-9. Correlation Between Sections in Chapter 3.0 of this Report and Appendixes in the Multi-Canister Overpack Design Report.

Appendix referenced	This report section
1	MCO drawings
2	3.2.5
3	3.1.4
4	3.2.4
5	3.2.2
6	3.2.4
7	3.2.3
8	3.2.3
9	3.2.3
10	3.2.4
11	3.2.1
12	Not included* (see Goldmann 2000a)
13	Not included* (see Goldmann 2000a)
14	Not included* (see Goldmann 2000a)
15	Not included* (see Goldmann 2000a)
16	Not included* (see Goldmann 2000a)
17	Not included* (see Goldmann 2000a)

Note: Appendixes cited are found in Goldmann, L. H., 2000a, *Multi-Canister Overpack Design Report, Book II - Calculations*, HNF-SD-SNF-DR-003, Rev. 3, Fluor Hanford, Incorporated, Richland, Washington.

*In general, the appendixes not included in this table do not contain structural data.

3.2 STRUCTURAL ANALYSES

The structural analyses for the normal operating loads and the accident conditions described in the MCO Performance Specification (Goldmann 2000b) have been conducted by the design agent, Parsons Infrastructure and Technology Group, Inc. Supplemental work on the effects of drops has been conducted by Hanford Site core organizations. The stress analyses for the MCO are described in the following subsections.

The materials used for the MCO pressure boundary are different grades of 304 stainless steel. However, all materials are purchased as dual certified. This means that all grades used (e.g., 304L) will have the higher properties of 304 stainless steel. In the attached analyses the properties and stress ratios sometimes reflect the 304 stainless steel. This does not reflect the dual-certified properties and is thereby a conservative presentation.

3.2.1 Multi-Canister Overpack Pressure Boundary Stress and Thermal Analyses

This section addresses only the pressure boundary vessel. Other components that form portions of the boundary are addressed in Section 3.2.4. The MCO assembly is a single-purpose SNF container that is capable of maintaining subcriticality at all times and of maintaining SNF confinement after being closed and sealed. The MCO assembly consists of a shell, bottom baseplate, shield plug, locking ring, jacking screws, and cover cap.

The MCO shell is fabricated of type 304L stainless steel with the minimum tensile and yield strengths of type 304 stainless steel (dual-certified 304/304L stainless steel). For the analysis, values for material properties and allowable stress values were taken from Section II, Part D, of the ASME Code (ASME 1998) (see Table 3-10). The canister cover is fabricated from SA-182 F304L stainless steel, the shield plug with SA-182 F304L stainless steel, the locking ring with SA-182 F304N stainless steel, jacking screws with ASME SA-193 grade B8S stainless steel, the collar with SA-182 dual certified 304/304L stainless steel, and the cover cap with SA-182 F304L stainless steel.

The calculations performed by the design agent considered thermal, lifting, and pressure loads (Goldmann 2000a, Appendix 11). The allowable stress intensities are those specified by Section III, Subsection NB-3220, of the ASME Code (ASME 1998). For normal condition loading, the MCO was analyzed according to service level A stress intensity limits (ASME 1998).

Table 3-10. Material Properties Defined in the American Society of Mechanical Engineers
Boiler and Pressure Vessel Code. (2 sheets)

SA-182 F304L Austenitic Stainless Steel Forging (Shield Plug)				
	70 °F	200 °F	300 °F	270 °F*
E (lb/in ²)	28.3 E+06	27.6 E+06	27.0 E+06	27.18 E+06
S _M (lb/in ²)	16,700	16,700	16,700	16,700
S _Y (lb/in ²)	25,000	21,300	19,100	19,760
S _U (lb/in ²)	65,000	61,500	56,500	58,000
Mean Coefficient of Thermal Expansion from 70° to Temp. - in/in/°F x 10 ⁻⁶				
	70 °F	250 °F	300 °F	270 °F
α-in/in/°F	8.46 E+06	8.90 E+06	9.00 E+06	8.94 E+06
SA-182 F304N Austenitic Stainless Steel Forging (Locking Ring)				
	70 °F	200 °F	300 °F	270 °F
E (lb/in ²)	28.3 E+06	27.6 E+06	27.0 E+06	27.18 E+06
S _M (lb/in ²)	23,300	23,300	22,500	22,740
S _Y (lb/in ²)	35,000	28,700	25,000	26,110
S _U (lb/in ²)	80,000	80,000	75,900	77,130
Mean Coefficient of Thermal Expansion from 70° to Temp. - in/in/°F x 10 ⁻⁶				
	70 °F	250 °F	300 °F	270 °F
α-in/in/°F	8.46 E+06	8.90 E+06	9.00 x 10 ⁶	8.94 x 10 ⁶
SA-182 F304 Stainless Steel Forging (Shell Collar)				
	70 °F	200 °F	300 °F	270 °F
E (lb/in ²)	28.3 x 10 ⁶	27.6 x 10 ⁶	27.0 x 10 ⁶	27.18 x 10 ⁶
S _M (lb/in ²)	20,000	20,000	20,000	20,000
S _Y (lb/in ²)	30,000	25,000	22,500	23,250
S _U (lb/in ²)	75,000	71,000	66,000	67,500
Mean Coefficient of Thermal Expansion from 70° to Temp. - in/in/°F x 10 ⁻⁶				
	70 °F	250 °F	300 °F	270 °F
α-in/in/°F	8.46 x 10 ⁶	8.90 x 10 ⁶	9.00 x 10 ⁶	8.94 x 10 ⁶

Table 3-10. Material Properties Defined in the American Society of Mechanical Engineers
Boiler and Pressure Vessel Code. (2 sheets)

SA-193 Grade B8S or B8SA (Bolting)				
	70 °F	200 °F	300 °F	270 °F
E (lb/in ²)	28.3 x 10 ⁶	27.6 x 10 ⁶	27.0 x 10 ⁶	27.18 x 10 ⁶
S _M (lb/in ²)	16,700	13,000	11,000	11,600
S _Y (lb/in ²)	50,000	--	--	--
S _U (lb/in ²)	95,000	--	--	--
Mean Coefficient of Thermal Expansion from 70° to Temp. - in/in/°F x 10 ⁻⁶				
	70 °F	250 °F	300 °F	270 °F
α-in/in/°F	8.46 x 10 ⁶	8.90 x 10 ⁶	9.00 x 10 ⁶	8.94 x 10 ⁶

*Values are interpolated from ASME Code values (ASME 1998).

E = elastic modulus.

S_M = code allowable stress.

S_U = material ultimate stress.

S_Y = material yield stress.

psi = pounds per square inch.

α-in/in/°F = thermal coefficient of expansion.

3.2.1.1 Shell Design. The MCO shell and bottom baseplate have been analyzed for internal pressure using classical methods (Goldmann 2000a, Appendix 11). The allowable external pressure for the shell has been calculated in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NB, Paragraph NB-3133.2 (ASME 1998). The design internal pressure of the MCO is 150 lb/in² gauge with the mechanical closure alone. With the cover cap welded in place, the design internal pressure is 450 lb/in² gauge. Pressure on the bottom of the MCO has an added component due to the weight of the fuel, approximately 16,000 lb (see Table 3-6).

The fuel is conservatively assumed to act as a fluid, resulting in lateral pressure against the shell walls. The added pressure is taken as that pressure on the bottom baseplate due to fuel weight and has been calculated to be 39 lb/in²; the total internal pressure used in the analysis is 189 lb/in².

The inside diameter of the MCO shell used in the analysis is 23.00 in. and the outer diameter is 24.00 in. The wall thickness is therefore 0.5 in. The stress through the shell due to the pressure load was calculated to be 4,442 lb/in².

During process operating conditions, the MCO is subjected to a full internal vacuum with a 25 lb/in² gauge external pressure; this is equivalent to external pressures of 14.7 lb/in² + 25 lb/in², or approximately 40 lb/in², at 75 °C (167 °F).

The allowable external pressure was calculated to be 181 lb/in². This value is greater than the 40 lb/in² maximum equivalent external pressure, therefore the cylindrical portion of the shell is adequate for external pressure.

3.2.1.2 Stress and Thermal Analysis. The stress analysis of the MCO assembly has been performed using the computer analysis program ANSYS⁵ (Swanson 1992). The ANSYS model is built using two-dimensional axisymmetric elements. To model the threads between the shell and locking ring, coincident nodes are coupled. Coupled nodes also are used to model the threads between the locking ring and the jacking screw. Symmetry boundary conditions are applied to all nodes along the centerline. The axisymmetric model used in this analysis is shown in Figures 3-2, 3-3, and 3-4.

Five process operating conditions have been analyzed for thermal and pressure loads on the MCO pressure vessel boundary:

1. Full internal vacuum with 60 lb/in² gauge external pressure; equivalent to external pressures of approximately 60 lb/in², at 75 °C uniform temperature (all stresses for this load case are classified as primary stresses [P_M or $P_L + P_B$])
2. Full internal vacuum with 0.0 lb/in² gauge external pressure; equivalent to external pressure of 14.7 lb/in², or approximately 15 lb/in², at 132 °C uniform temperature (all stresses for this load case are classified as primary stresses)
3. Internal pressure of 189 lb/in² at 132 °C uniform temperature, which represents 150 lb/in² for the internal pressure and 39 lb/in² for the fuel weight (all stresses for this load case are classified as primary stresses)
4. Lifting the MCO with 450 lb/in² internal pressure and 39 lb/in² for the fuel weight at 132 °C uniform temperature (all stresses for this load case are classified as primary stresses).
5. Differential temperature, shell at 132 °C and shield plug at 32 °C, at 189 lb/in² internal pressure (all stresses for this load case are classified as primary plus secondary [$P_L + P_B + Q$] because thermal stresses are secondary stresses [Q]; the primary stresses for this load case are the same as for load case 1).

⁵ANSYS is a trademarked proprietary code, but a users license is commercially available from Swanson Analysis Systems, Inc., Houston, Pennsylvania.

A summary of the maximum stress intensities is presented in Tables 3-11 through 3-15. For load cases 1, 2, 3, and 4 (internal and external pressures), the primary membrane stress, P_M , is compared to the allowable membrane stress, S_M ; the membrane plus bending stress, $P_L + P_B$, is compared to $1.5 S_M$. For load case 5 (differential temperature), the total stress, $P_L + P_B + Q$, is compared to $3 S_M$. The results show that for all load cases, the computed stress intensities are lower than the allowable stress intensities.

3.2.2 Multi-Canister Overpack Buckling and Drop-Related Analyses

This section discusses the design agent's evaluation of the MCO assembly under different drop loads. The evaluation is based on criteria in the ASME Boiler and Pressure Vessel Code (ASME 1998). The following assumptions have been used in the evaluation.

- Pressure is applied uniformly.
- The maximum weight of a fully loaded Mark IV basket is 3,218.52 lb. The analysis is conservative, using 3,225 lb for each basket.
- Maximum shell diameter at the collar increases from 25.27 to 25.31 in. Calculations performed in this analysis are based on 25.27 in. and, therefore, are conservative.
- Other assumptions are applied as noted. This section considers only drop loads. The evaluated drop loads must meet Section III, Subsection NB, of the ASME Boiler and Pressure Vessel Code, with Appendix F (ASME 1998), and Section 4.9.5 of the MCO Performance Specification (Goldmann 2000b). One of the acceptance criteria in Appendix F (F-1331.1) of the ASME Code (ASME 1998) states that when an elastic analysis is used "the general primary membrane stress intensity P_M shall not exceed the lesser of $2.4 S_M$ and $0.7 S_U$ for materials included in Section II, Part D, Subpart I, Tables 2A and 2B for austenitic and high alloy steels, or $0.7 S_U$ for ferritic materials included in Table 2A," and "the local primary membrane stress intensity P_L shall not exceed 150% of the limit for general primary membrane stress intensity P_M ." Section 4.9.5 of the MCO Performance Specification (Goldmann 2000b) states that "In no case, including post accident conditions, is the MCO inside circumference below the bottom of the shield plug allowed to exceed 73.04 inches (23.25 inches x π)." Ultimately the MCO shell inside diameter is not to exceed 23.25 in.
- The collar thread relief thickness is conservatively modeled at 0.354 in.

Table 3-11. Summary of Maximum Stress Intensities for Load Case 1
(Full Internal Vacuum with External Pressure of 60 lb/in² at 75 °C).

Component	Material	S_M (ksi)	P_M (ksi)	Stress ratio	$P_L + P_B$ (ksi)	Stress ratio
Bottom plate	304L	16.7	0.94	0.06	2.56	0.10
Lower shell	304*	20.0	1.02	0.05	1.88	0.07
Middle shell	304*	20.0	1.01	0.05	1.03	0.03
Upper shell	304*	20.0	6.26	0.31	9.6	0.32
Shield plug	304L	16.7	6.17	0.37	7.08	0.28
Locking ring	304N	23.3	1.08	0.05	1.55	0.04

*Dual certified 304/304L.

$$\text{NOTE: Stress ratio} = \frac{P_M}{S_m} \text{ or } \frac{P_L + P_B}{1.5 S_M} \text{ with } S_m \text{ at } 75^\circ \text{C (167 }^\circ \text{F)}.$$

S_M = allowable membrane stress.

P_M = primary membrane stress.

$P_L + P_B$ = membrane plus bending stress.

Table 3-12. Summary of Maximum Stress Intensities for Load Case 2
(Full Internal Vacuum with External Pressure of 0.0 lb/in² at 132 °C).

Component	Material	S_M (ksi)	P_M (ksi)	Stress ratio	$P_L + P_B$ (ksi)	Stress ratio
Bottom plate	304L	16.7	0.35	0.02	0.96	0.04
Lower shell	304*	20.0	0.38	0.02	0.71	0.02
Middle shell	304*	20.0	0.38	0.02	0.39	0.01
Upper shell	304*	20.0	5.61	0.28	8.87	0.30
Shield plug	304L	16.7	7.2	0.38	7.05	0.28
Locking ring	304N	22.74	1.14	0.05	1.62	0.04

*Dual certified 304/304L.

$$\text{NOTE: Stress ratio} = \frac{P_M}{S_m} \text{ or } \frac{P_L + P_B}{1.5 S_M} \text{ with } S_m \text{ at } 132^\circ \text{C (270 }^\circ \text{F)}.$$

S_M = allowable membrane stress.

P_M = primary membrane stress.

$P_L + P_B$ = membrane plus bending stress.

Table 3-13. Summary of Maximum Stress Intensities for Load Case 3
(Internal Pressure of 189 lb/in² at 132 °C).

Component	Material	S _M (ksi)	P _M (ksi)	Stress ratio	P _L + P _B (ksi)	Stress ratio
Bottom plate	304L	16.7	2.83	0.17	9.35	0.37
Lower shell	304*	20.0	4.6	0.23	10.72	0.36
Middle shell	304*	20.0	4.62	0.23	4.71	0.16
Upper shell	304*	20.0	5.04	0.25	8.4	0.28
Shield plug	304L	16.7	5.2	0.31	7.2	0.29
Locking ring	304N	22.74	1.37	0.06	1.93	0.06

*Dual certified 304L/304.

NOTE: Stress ratio = $\frac{P_M}{S_m}$ or $\frac{P_L + P_B}{1.5S_M}$ with S_m at 132 °C (270 °F).

S_M = allowable membrane stress.

P_M = primary membrane stress.

P_L + P_B = membrane plus bending stress.

Table 3-14. Summary of Maximum Stress Intensities for Load Case 4
(Lifting with Internal Pressure of 450 lb/in² at 132 °C).

Component	Material	S _M (ksi)	P _M (ksi)	Stress ratio	P _L + P _B (ksi)	Stress ratio
Bottom plate	304L	16.7	7.41	0.44	24.76	0.98
Lower shell	304*	20.0	11.92	0.60	18.2	0.61
Middle shell	304*	20.0	11.96	0.60	12.19	0.41
Upper shell	304*	20.0	N/A	N/A	16.3	0.54
Shield plug at 150 lb/in ² gauge	304L	16.7	4.84	0.29	6.95	0.28
Locking ring at 150 lb/in ² gauge	304N	22.74	1.62	0.07	3.02	0.09

*Dual certified 304L/304.

NOTE: Stress ratio = $\frac{P_M}{S_m}$ or $\frac{P_L + P_B}{1.5S_M}$ with S_m at 132 °C (270 °F).

S_M = allowable membrane stress.

P_M = primary membrane stress.

P_L + P_B = membrane plus bending stress.

Table 3-15. Summary of Maximum Stress Intensities for Load Case 5
(Differential Temperature — Internal Pressure of 189 lb/in²
with Shell at 132 °C and Shield Plug at 32 °C).

Component	Material	S _M (ksi)	Differential temperature	
			P _L + P _B + Q	Stress ratio
Bottom plate	304L	16.7	26.66	0.53
Lower shell	304*	20.0	9.65	0.16
Middle shell	304*	20.0	4.81	0.08
Upper shell	304*	20.0	8.99	0.15
Shield plug	304L	16.7	14.2	0.28
Locking ring	304N	22.74	2.54	0.04

*Dual certified 304L/304.

NOTE: Stress ratio = $\frac{P_L + P_B + Q}{3S_m}$ with S_m at 132 °C for shell, rest at 32 °C.

S_M = allowable membrane stress.

P_L + P_B + Q = total stress.

3.2.2.1 Filter Guard Plate and Support Ring. The guard plate on the MCO shield plug has been evaluated for its ability to protect the internal filters in a top-down drop. The guard plate and its support ring are fabricated from SA-240 type 304L stainless steel. The support ring is a 1-in.-thick ring, 4.25 in. high, with a 22.50-in. outer diameter. The guard plate must withstand a 28 g load of the entire payload of the MCO, including baskets, without deflecting more than 1 in. A load level of 28 g bounds the loading case of 27.4 g (lid down) in Table 3 of the MCO Performance Specification (Goldmann 2000b). To be conservative, no credit is taken for the baskets supporting any of the fuel. Service level D allowable stresses are used for the acceptance criteria.

HNF-SD-SNF-DR-003, Appendix 5 (Goldmann 2000a), provides calculations to assess the buckling of the guard plate support ring and to size the weld between the support ring and the shield plug and the weld between the guard plate and the support ring. The assumptions used in the calculations are that the fuel loads the plate uniformly, the baskets provide structural capacity to distribute the fuel load, the temperature at impact is 132 °C, and the maximum weight results from five Mark IV baskets (each basket weighs a maximum of 3,225 lb).

The load for the buckling check of the support ring is an acceleration of the fuel at 28 g. For the welds, an acceleration of 101 g is used as it provides a worst-case acceleration for a side load.

Analysis of the support ring shows the stress is well below allowable levels and that buckling is not a concern. The stress in the support ring when assuming that the ring supports the entire load is 6,685 lb/in², which is well below the service level D allowable levels shown in Table 3-16. The deflection was calculated to be 0.001 in., which is well below the 1.0 in. allowed.

Table 3-16. Service Level D Allowable Stresses for Elastic Analysis
at a Temperature of 132 °C.

Material	S _M (ksi)	Service level D stress limits* (ksi)		
		P _m (2.4S _M)	P _L (1.5P _m)	P _M (or P _L) + P _B (1.5P _m)
Type 304	20.0	47.25 (0.7S _U)	67.5	67.5
Type 304L	16.7	40.08	60.12	60.12

*Service level D stress limits are taken from Appendix F, Paragraph F-1331, of the 1998 edition of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code*.

Welding the support ring to the shield plug and the guard plate to the support ring requires the use of structural welds. The minimum weld size required, from calculations in HNF-SD-SNF-DR-003 (Goldmann 2000a, Appendix 5) is

- Throat weld = 0.016 in.
- Base weld = 0.028 in.

However, the minimum allowed weld for joining plate material of this thickness is 0.3125 in. (AISC 1989, Table 1.17.2a). The stress ratio in the welds at 0.3125 in. are very low.

3.2.2.2 Leakage of the Closure Seal. The possibility that the closure seal could leak after a DBA drop has been reviewed. This DBA drop scenario assumes that the MCO at the bottom of the CSB storage tube has not had a cover cap welded in place over the shield plug and that no intermediate impact absorber has been put in place before the top MCO is loaded into the tube. A drop of one MCO onto another MCO inside the CSB tube is defined by the MCO Performance Specification as a 28 g load (Goldmann 2000b). This acceleration was determined and recorded in HNF-SD-SNF-DP-007 (Chenault 1997a, Appendix D) for the drop of an MCO onto another without an intermediate impact absorber.

This drop load impacts the top of the bottom MCO shield plug, which, in turn, reacts through the closure seal–seal stop at the bottom of the shield plug. HNF-SD-SNF-DR-003, Appendix 5 (Goldmann 2000a), states that the seal stop does not exceed the yield for this case when the MCO has 150 lb/in² gauge internal pressure. The design agent has shown that the seal stop strain exceeds yield for the worst case of zero pressure during a drop accident. Calculations

performed by the design agent show that the seal deformation is within the allowable compression and that the seal will still maintain a leaktight seal (given as 0.012 in. beyond normal compression). In addition, the likelihood of dropping an MCO without the cover cap is small, as is the likelihood of dropping an MCO without an intermediate impact absorber in place.

3.2.2.3 Multi-Canister Overpack Assembly. The MCO assembly has been evaluated for the following drops:

- 54 g bottom end drop, 450 lb/in² internal pressure, 132 °C
- 28 g drop of a fully loaded MCO onto another MCO without cover cap
- 28 g drop of a fully loaded MCO onto another MCO with cover cap.

These drops have been analyzed by both engineering mechanics and finite element methods. The results of the finite element methods are described in the following sections. The finite element evaluation of the MCO assembly was performed using the axisymmetric (two-dimensional) model shown in Figures 3-5 and 3-6. The model was developed using the ANSYS computer code (Swanson 1992). Symmetry boundary conditions are applied at the edges of the model. Nodes between the set screws (called "jacking bolts" in the figures) and the locking ring, and nodes between locking ring threads and shell threads are coupled (see Figure 3-3). The gap elements between the shield plug and the bottoms of the set screws have a very small positive interference to represent the preload on the screws.

3.2.2.3.1 Bottom End Drop. HNF-SD-SNF-DR-003, Appendix 5, Section 9.0, "MCO Structural Drop Analysis" (Goldmann 2000a), provides an evaluation for a bottom end drop. The MCO shell and its contents have been evaluated for a 2-ft, 54 g, vertical drop onto a flat reinforced concrete surface. The MCO lands on the bottom end and there is no credible possibility of a side-slap-down secondary impact of the MCO.

The analysis assumes the drop occurs when the MCO is at 132 °C with an internal pressure of 450 lb/in². It also is assumed that the MCO is fully loaded with five Mark IV baskets, two of which are scrap baskets. The fuel is modeled with the properties of stainless steel except for the scrap baskets, which have hydrostatic properties when externally loaded. No credit is taken for the baskets' ability to support the fuel. Upon impact it is assumed that the bottom basket is crushed. The pressure acting on the top of the bottom basket is equivalent to the weight of the four baskets above it and is calculated to be 503 lb/in² for an MCO internal area of 415.48 in².

The fuel exerts a horizontal pressure equal to the vertical pressure against the wall of the shell. At any level, the vertical pressure equals the weight of a 1-ft² column of fuel above that level. The horizontal pressure was calculated, a triangular distribution assumed, resulting in additional pressure. The maximum pressure at the base of the triangular distribution would be 1,051 lb/in². This pressure is in addition to the internal 450 lb/in² stated above.

According to Section 4.9.5 of the MCO Performance Specification (Goldmann 2000b), the shell's maximum circumference is not to exceed 73.04 in. (23.25 in. $\times$ π). At 23.25 in., the maximum radial displacement allowed is 0.25 in. diametrically, or 0.125 in. radially. The radial

displacement at a point above the inside bottom baseplate of the MCO was calculated using examples from *Equations for Stress and Strain* (Roark 1989, Table 28) for deformation in a thin-walled pressure vessel with a uniform radial pressure and an additional, varying, linearly radial pressure. A combination of Case 1b and Case 1d from Roark (1989) was used. The calculation results in a total radial displacement for a bottom end drop of 0.0044 in., which is well below the maximum of 0.125 in. allowed.

3.2.2.3.2 Multi-Canister Overpack onto Multi-Canister Overpack Drop. A 28 g vertical drop of one MCO onto another MCO in the storage tube at the CSB is analyzed in HNF-SD-SNF-DR-003 (Goldmann 2000a). The top of the lower MCO being impacted receives a 28 g vertical load and is evaluated with and without the cover cap welded in place.

The drop is assumed to occur with the MCO at 132 °C (270 °F) with an internal pressure of 450 lb/in². In the analysis, it is assumed that the MCO impacting the lower MCO is fully loaded with five Mark IV baskets. The lower MCO is restrained by the CSB storage tube. The effect on the lower MCO is a concern because the shell is subject to buckling. The top of the MCO being impacted receives the equivalent of a 28 g load from the upper MCO landing on the shield plug or the cover cap of the lower MCO, depending on the evaluation.

The equivalent pressure received by the top of the lower MCO shield plug is calculated by taking the weight of a fully loaded, dry MCO, multiplied by the gravitational acceleration and divided by the area of the shield plug or the cover cap.

If the MCO being impacted does not have the lifting cap, it is assumed that all the weight hits the shield plug first and not the locking ring as the shield plug extends 0.125 in. above the top of the locking ring. From Table 3-6, the weight of a dry MCO containing the upper shield plug and five loaded Mark IV baskets is 19,576 lb. The area computed for the shield plug is 39.54 in². The calculation results in a stress of nearly 13,700 lb/in².

If the MCO being impacted has the cover cap welded in place, it is assumed that all the weight is distributed evenly. That distribution is represented as an equivalent pressure acting on top of the cover cap with a computed area of 316.6 in². The calculation results in a stress of 15,172 lb/in². Buckling of the shell is evaluated using the ANSYS program.

3.2.2.3.3 Finite Element Analysis for Drops. An evaluation of the MCO assembly has been performed using a finite element model. The model is axisymmetric (two-dimensional) and was developed with elements having 2 degrees of freedom at each node.

Symmetry boundary conditions are applied at the edges of the model. Nodes between the jacking bolts and the locking ring, and nodes between locking ring threads and shell threads, are coupled. The gap elements between the shield plug and the bottom of the jacking bolts have a very small positive interference to represent the preload on the bolts.

In the bottom end drop (54 g load), gap elements are used at the bottom of the model in order to anchor the model. In the MCO onto MCO drop (28 g equivalent), the load is applied at

the top of the shield plug or at the top of the lifting cap, and gap elements at the bottom of the model are also restrained in both X and Y directions.

Bottom End Drops. Two load cases were used to analyze bottom end drops. One load case analyzed the 54 g bottom end drop of an MCO with 150 lb/in² internal pressure and without the lifting cap. The second load case used the same model to analyze the 54 g bottom end drop of an MCO with 450 lb/in² internal pressure and with the lifting cap. The lifting cap was not explicitly modeled for the second case. The load from the weight of the cap was applied to the top of the canister collar.

The bottom drop analysis was performed with hydrostatic pressure in addition to the design pressures (150 lb/in² and 450 lb/in²). The load applied by the payload was modeled as a pressure exerted on the bottom plate of the MCO equivalent to the weight of five loaded Mark IV baskets distributed over the area of the plate:

$$A_p = \text{MCO bottom plate area} = \pi \times \frac{(23.00)^2}{4} = 415.48 \text{ inch}^2$$

$$W_B = \text{Maximum weight of a Mark IV basket} = 3,218 \text{ lb}$$

$$P_{bp} = \frac{5(W_B)(54 \text{ g})}{A_p} = \frac{5(3,218)(54 \text{ g})}{415.48^2} = 2,090 \text{ lb/inch}^2$$

The geometry of the MCO, storage tube, and MHM is being designed to prevent large off-center loading or dropping. Analyses have been conducted (HNF-SD-SNF-DP-007, Appendix C) that show small offsets are acceptable when striking an edge. This shows the weight distribution assumption was reasonable. Further analyses are being conducted with the fuel weight distributed as actually configured and with larger offsets; preliminary results are showing the imposed strains are acceptable.

The hydrostatic pressure described in Section 3.2.2.3.1 is applied, except that the applied hydrostatic pressure is based on the earlier maximum basket weight of 3,200 lb rather than a bounding weight of 3,225 lb. Although not conservative, the payloads applied to the finite element analyses are based on 3,200 lb. The difference is less than 1%, and considering the low stress ratios, the effect is negligible.

Symmetry boundary conditions are applied at the edges of the model. Nodes between the jacking bolts and the locking ring, and nodes between locking ring threads and shell threads are coupled. The gap elements between the shield plug and the bottom of the jacking bolts have a very small positive interference to represent the preload on the bolts. Gap elements are used at the bottom of the model in order to simulate contact between the bottom plate and ground while allowing for bending and/or axial distortion of the bottom plate.

A summary of the maximum stress intensities and stress ratios for both load cases is provided in Table 3-17.

Table 3-17. Summary of Maximum Stress Intensities for Bottom End Drop (54 g).

Component	Membrane stress intensities		Membrane + bending stress intensities	
	Maximum (lb/in ²)	Stress ratio	Maximum (lb/in ²)	Stress ratio
Load case at 150 lb/in ² internal pressure without lifting cap				
Bottom plate	3,344	0.07	7,161	0.11
Lower shell	19,930	0.42	22,310	0.33
Middle shell	5,735	0.12	5,815	0.09
Upper shell and collar	4,604	0.10	7,239	0.11
Shield plug	7,322	0.18	9,016	0.15
Guard plate	312.9	0.01	700.2	0.01
Locking ring	1,727	0.04	2,229	0.03
Load case at 450 lb/in ² internal pressure with lifting cap				
Bottom plate	3,933	0.08	8,441	0.13
Lower shell	25,060	0.53	28,440	0.42
Middle shell	10,710	0.23	10,950	0.16
Upper shell and collar	10,660	0.33	11,160	0.24
Shield plug	5,749	0.14	8,296	0.14
Guard plate	585	0.01	1,057	0.02
Locking ring	2,099	0.05	2,703	0.04

Multi-Canister Overpack onto Multi-Canister Overpack Drops. Two drops were analyzed: one with the cover cap welded in place and the other without. Both drops were at 132 °C. The drop at 150 lb/in² internal pressure was without the cover cap, and the drop at 450 lb/in² gauge was with the cover cap.

Analyses show that ASME Code allowables (Subsection NB and Appendix F) are not violated. The radial displacements due to the load cases and the pressure distribution are well within the allowables given by the MCO Performance Specification (Goldmann 2000b).

The results are summarized in Tables 3-18 through 3-21. The stress ratios in Tables 3-19 and 3-20 are calculated as the ratio of the maximum stress intensity to the allowed stress for that type of stress intensity. From Table 3-18, it is apparent that the maximum internal deformed shell circumference will not exceed the allowed value of 73.04 in. specified in the MCO Performance Specification (Goldmann 2000b).

3.2.2.3.4 End Drop Buckling. The maximum compressive membrane stress has been checked according to Paragraph NB-3133.6 of the ASME Boiler and Pressure Vessel Code (ASME 1998). From the analyses using the bottom end drop models, the maximum compressive axial stress in the MCO shell occurring during the postulated drops is 15,620 lb/in². This result occurs during the drop of one MCO onto another MCO at the CSB, with the end cap and zero internal pressure. According to Paragraph NB-3133.6(b) (ASME 1998), parameter A is

$$A = \frac{0.125}{\frac{R_c}{T}}$$

where

R_c = the inside radius of the MCO shell = 11.50 in.

T = the minimum thickness of the MCO shell = 0.485 in.

$$A = \frac{0.125}{\left(\frac{11.50}{0.485} \right)} = 0.0052$$

The corresponding value of B, obtained from the ASME Boiler and Pressure Vessel Code, Section II, HA-1 (ASME 1998), at 132 °C (270 °F) is 11,300 lb/in² (interpolated value). According the Boiler and Pressure Vessel Code, Paragraph F-1331.5, the allowable compressive stress is equal to 150% of B, or 16,950 lb/in². Since this value is greater than the computed compressive stress, the MCO shell meets the buckling criterion for an end drop. Although Table 3-17 reports maximum stress intensities greater than 16,950 lb/in², none of the compressive axial stresses exceed this value.

Table 3-18. Maximum Shell Radial Displacement and Circumference Change.

	54 g bottom end drop without lifting cap	54 g bottom end drop with lifting cap	28 g drop of one MCO onto another without lifting cap	28 g drop of one MCO onto another with lifting cap
Maximum radial displacement	0.0292 in.	0.0320 in.	0.0250 in.	0.0481 in.
Undeformed shell diameter	23.00 in.	23.00 in.	23.00 in.	23.00 in.
Maximum deformed shell diameter	23.0584 in.	23.0640 in.	23.0500 in.	23.0962 in.
Maximum calculated circumference	72.44 in.	72.46 in.	72.41 in.	72.56 in.
Allowed circumference	73.04 in.	73.04 in.	73.04 in.	73.04 in.

MCO = multi-canister overpack.

Table 3-19. Summary of Maximum Stress Intensities for Multi-Canister Overpack onto Multi-Canister Overpack Drop without Cover Cap in Place (28 g).

Component	Membrane stress intensities		Membrane + bending stress intensities	
	Maximum (lb/in ²)	Stress ratio	Maximum (lb/in ²)	Stress ratio
Bottom plate	8,522	0.18	12,030	0.18
Lower shell	16,930	0.36	19,730	0.29
Middle shell	16,450	0.35	16,520	0.24
Upper shell and collar	16,630	0.35	17,760	0.26
Shield plug	19,040	0.48	21,740	0.36
Guard plate	5,776	0.14	9,496	0.16
Locking ring	977	0.02	1,013	0.02

Table 3-20. Summary of Maximum Stress Intensities for Multi-Canister Overpack onto Multi-Canister Overpack Drop with Cover Cap in Place (28 g).

Component	Membrane stress intensities		Membrane + bending stress intensities	
	Maximum (lb/in ²)	Stress ratio	Maximum (lb/in ²)	Stress ratio
Bottom plate	11,850	0.25	19,040	0.28
Lower shell	21,170	0.45	22,890	0.34
Middle shell	20,640	0.44	20,850	0.31
Upper shell and collar	29,270	0.92	39,080	0.86
Shield plug	1,477	0.04	3,230	0.05
Guard plate	1,799	0.04	2,277	0.04
Locking ring	8,373	0.21	12,990	0.22
Lifting cap	27,880	0.70	33,180	0.55

Table 3-21. Summary of Guard Plate Maximum Deflections.

Load case	Maximum plate deflection
54 g bottom end drop, without cover cap	0.008
54 g bottom end drop, with cover cap	0.008
28 g drop into CSB tube without cover cap	0.007
28 g drop into CSB tube with cover cap	0.008

CSB = Canister Storage Building.

3.2.3 Multi-Canister Overpack Fuel Storage Basket Analyses

The following sections summarize the design agent's evaluation of the Mark IA and Mark IV storage and scrap baskets for lifting and deadweight loading. The Mark IA storage and scrap baskets also have been evaluated for drop accident loading. The structural adequacy evaluation is based on meeting the safety factors of three on yield and five on ultimate strength as listed in the MCO Performance Specification (Goldmann 2000b) for lifting and loading and on the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME 1998), for the deadweight stacking and drop load conditions.

3.2.3.1 Stress Analysis of the Mark IA Fuel and Scrap Baskets. This section summarizes the design agent's evaluation of the Mark IA fuel and scrap baskets for lifting, deadweight, and drop accident loading (Goldmann 2000a, Appendix 7). The structural components of the Mark IA fuel and scrap baskets are identical.

The following assumptions were used in the evaluation.

- For the vertical drop loading when the baskets are stacked within the MCO, it is conservatively assumed that the basket support rods are in alignment, with the exception of the bottom basket, which is assumed to be rotated 30° relative to the baskets above. This configuration produces the maximum bending in the basket bottom plate.
- Because the MCO Performance Specification (Goldmann 2000b) does not specify the density of the scrap material in a loaded scrap basket, density is calculated based upon the assumption that the total basket scrap weight is equal to the total fuel weight (48 fuel rods) in a loaded Mark IA storage basket.
- For the horizontal drop evaluation, the top end support for the center pipe, which interfaces with either the basket above or the bottom of the shield plug assembly, must be maintained throughout the drop.

Other assumptions are as noted within the calculation documentation.

The geometry of the Mark IA fuel and scrap baskets is defined in Figures 1-5 and 1-7, respectively. The primary structural components for both basket designs (bottom plate, support rods, and center pipe) are identical. The primary structural components and other significant components of the fuel basket are identified in Figure 3-7. The scrap basket has additional components in the form of six 60° shroud segments. Each segment is fabricated from copper plate. When all six segments are fastened together they provide a fines area around the center pipe and stiffener plates between the outer shroud and the fines area. The shrouds provide no structural function except to position the scrap material during initial loading.

A summary of the function of each structural component is provided in Table 3-22. The bottom plate has a thickness of 1.2 in. with 0.5-in. flow holes, except in the center, where the bottom plate is attached to the center post, the thickness is 1.5 in. to accommodate the threaded connection. The center post is fabricated from 6.625-in.-outer-diameter bar stock and fastened to the bottom plate using 5-in.-diameter, 4-pitch acme stub threads. The top of the center post features a center coupling that interfaces with the lifting grapple and also provides stability by interfacing with a mating provision in the basket above when the baskets are stacked within the MCO. The coupling on the uppermost basket interfaces with the shield plug assembly. The approximately trapezoidal geometry of the support rods provides the maximum cross section for the available space. A thick aluminum fuel-positioning plate, or fuel rack (storage basket only), rests on the bottom plate. The fuel rack serves a locational function and was not subjected to a structural evaluation.

Table 3-22. Mark IA Fuel Basket Structural Components.

Component name	Structural functions
Center pipe	(1) Primary load-carrying component during lifting operations (2) Provides support to baskets above when stacked inside the MCO (3) Provides "void space" boundary for criticality safety, which must be maintained during normal operations and following drop accidents ^a
Center coupling	(1) Primary load-bearing component during lifting operations (2) Shear support for the center pipe during a horizontal drop accident
Support rod	Provides support to baskets above when stacked inside the MCO for normal operations and during vertical drop accidents
Bottom plate	(1) Supports the fuel during normal operations (2) Maintains the position of the center pipe during drop accidents

^aGoldmann, L. H., 2000b, *Performance Specification for the Spent Nuclear Fuel Multi-Canister Overpack*, HNF-S-0426, Rev. 6, Fluor Hanford, Incorporated, Richland, Washington.

Both the storage and scrap basket designs include a shroud located on the outside circumference of the baskets. The storage basket shroud is fabricated out of 18-gauge sheet metal and is half-height. The scrap basket shroud is fabricated out of copper and is full-height. The use of copper for the scrap basket shroud increases dissipation of the heat produced by the fuel and is a modification from the original design. Since the fuel rack in the storage basket prevents significant loading to the shroud, the storage basket shroud is considered to be nonstructural. The scrap basket shroud is fabricated out of copper, so it is not classified as a structural component. However, since the scrap basket shroud is subjected to a relatively low pressure loading from the scrap pieces bearing against it, an evaluation has been performed to demonstrate copper's behavior under pressure and elevated temperature.

The SPR fuel basket is a modified version of the Mark IA fuel basket. A nonstructural loading jig replaces the aluminum fuel rack. The fuel elements are stacked two to three high in each loading position. The combined weight of the jig and the SPR fuel in the Mark IA fuel basket is less than the weight of the Mark IA fuel. Therefore all analyses for the Mark IA basket are bounded by the Mark IA fuel load.

The bottom plate and center post are fabricated from 304L stainless steel, and the support rods are fabricated from dual-certified 304L/304 stainless steel. Other materials include the nonstructural aluminum fuel rack for the storage basket and the copper shroud segments for the scrap basket. For this analysis, the only mechanical properties of interest are the elastic modulus, yield strength, ultimate strength, and ASME stress allowable. At 132 °C, the yield strength of the copper shroud material is 7,650 lb/in² and ultimate strength is 30,000 lb/in².

3.2.3.1.1 Lifting Loads. According to Section 4.12.3 of the MCO Performance Specification (Goldmann 2000b), the Mark IA baskets shall meet the safety factors of three on yield and five on ultimate material strength. The safety factors apply from 5 °C to 100 °C (41 °F to 212 °F). These safety factors are to ensure the following:

"The load bearing members of a special lifting device shall be capable of lifting three times the combined weight of the shipping container with which it will be used, plus the weight of intervening components of the special lifting device, without generating a combined shear stress or maximum tensile stress at any point in the device in excess of the corresponding minimum tensile yield strength of their materials of construction. They shall also be capable of lifting five times that weight without exceeding the ultimate tensile strength of the materials."

The design criteria are to ensure that the shear stress shall be taken as an average value over the cross section and that the tensile stress may be due to direct or bending loads. The bending stress is defined as being linear over the cross section. These are the guidelines for which the baskets were designed. The "load bearing members of a special lifting device" are interpreted to apply to all components of the storage baskets in the load path between the lifting grapple and fuel.

3.2.3.1.2 Deadweight Loads. According to Section 4.12.1 of the MCO Performance Specification (Goldmann 2000b), the Mark IA baskets are to be designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME 1998). The deadweight stacking basket configuration is identical to the vertical drop accident configuration. Because the loading differences far exceed the allowable differences, the vertical drop accident condition obviously bounds the deadweight condition.

3.2.3.1.3 Drop Loading Conditions. The bounding vertical drop loading is 35 g and the horizontal drop loading is 101 g (see Table 3-1). The only potential sequential drop (vertical followed by horizontal) in the MCO Performance Specification (Goldmann 2000b) is the "Dropped with Cask" case, which specifies a 33.5 g vertical (corner) drop followed by a 101 g horizontal drop. In this report, the sequential drop is conservatively evaluated as a 35 g/101 g combination. A maximum drop temperature of 132 °C (270 °F) is specified in the MCO Performance Specification (Goldmann 2000b). The design agent's analyses were performed using a maximum drop temperature of 200 °C (392 °F). Because the analyses are conservative, they have not all been reanalyzed using a 132 °C (270 °F) temperature. The decrease in drop temperature has a significant influence on the allowable limits, particularly for allowable limits that are a function of the ultimate strength. For service level D events, the acceptance criteria are specified in Appendix F, Paragraph F-1440, which refers to Paragraph F-1300, of the ASME Boiler and Pressure Vessel Code (ASME 1998) with some specified exceptions.

In addition to the allowable stress limits, Section 4.19.3 of the MCO Performance Specification (Goldmann 2000b) stipulates a nuclear criticality safety requirement that a nominal void, 6.625 in. in diameter, be maintained at the basket centerline. For all load conditions,

including the drop accidents, this centerline void cannot deviate from the MCO centerline by more than 2 in.

According to the MCO Performance Specification (Goldmann 2000b), the shroud segment to bottom plate weld criterion is taken as

"the copper subassembly of the scrap basket shall be designed to withstand a distributed load in tension on the outside shroud of 10,350 pounds before yielding and 17,250 pounds before failure. This provides a safety factor of three to yield and five to failure during loading of the basket into the MCO."

This evaluation is shown in the MCO Design Report (Goldmann 2000a, Appendix 7).

3.2.3.1.4 Stress Analysis Calculations. The Mark IA storage baskets have been evaluated using both hand calculations and ANSYS finite element calculations. The finite element calculations were limited to stress predictions for the support rods and the relatively complex bottom plate.

Center Post. The center post and its connection to the bottom plate was evaluated for lifting and for both a 35 g vertical drop and a 101 g horizontal drop. For the vertical drop, the controlling limit is axial compression (column buckling), and for the horizontal drop, the threaded joint controls.

Lifting loads are carried by the center coupling and by the acme threaded connection between the center post and the bottom plate. The threads are seated with a torque of 340 to 400 ft-lb and fitted with a coiled spring pin to prevent loosening of the threaded connection. Since this torque is very small compared to the size of the threads, no further consideration of the torque is necessary. The preload of the center post thread is computed to equal the weight of the loaded basket times two to account for suddenly applied loads.

$$\text{Torque} = \frac{(0.17)(2)(2,400)(5.0)}{(12)} = 340 \text{ ft-lb (therefore use 340 to 400 ft-lb).}$$

The shear area of the acme threads is $A_s = 7.977 \text{ in}^2$ for an effective length of thread engagement of 1 in., or $(0.9)(7.977) = 7.179 \text{ in}^2$ for an effective thread length of 0.9 in. Conservatively assuming that the load on the joint is the entire weight of the basket, 2,400 lb, the shear stress in the threads is

$$\tau = \frac{2,400}{7.179} = 334 \frac{\text{lb}}{\text{in}^2}$$

The allowable stress in lifting is 7,010 lb/in². The stress ratio on thread shear for the acme threads is therefore

$$\text{Ratio} = \frac{334}{7,010 (0.6)} = 0.079$$

where the factor of 0.6 modifies the allowable for shear stress. The minimum diameter in the thread relief (d_{eff}) is 4.60 in. With the bushing bore diameter (d_b) of 2.71 in., the tensile stress area is

$$A_t = \frac{\pi}{4} (d_{\text{eff}}^2 - d_b^2) = 10.85 \text{ in}^2 .$$

The tensile stress is

$$\sigma = \frac{2,400}{A_t} = 221 \frac{\text{lb}}{\text{in}^2}$$

and the stress ratio is

$$\text{Ratio} = \frac{221}{7,010} = 0.032 .$$

Therefore, the center post attachment to the bottom plate is acceptable for the lifting condition.

For the vertical drop loading condition, the center post is loaded in compression. The lowest of the six baskets is subjected to the greatest load. The Mark IA basket weight is bounded by the weight of the storage basket at 2,400 lb. The buckling capacity of the center post was checked using the ASME Boiler and Pressure Vessel Code, Appendix F, Subsection F-1334.3(b) (ASME 1998) and the following input parameters. Material temperature is 132 °C.

$E = 27.2 \times 10^6 \text{ lb/in}^2$	304L stainless steel modulus
$S_y = 19.8 \text{ ksi}$	304L stainless steel yield strength
$K_p = 0.8$	Center post effective length factor, conservatively assumed pinned at top (Table C-C2.1 [AISC 1989])
$L_p = 22 \text{ in.}$	Length of center post, from threaded joint to flat top (bounding)
$R = 6.625/2 = 3.313 \text{ in.}$	Outside radius of center post

$$R_i = 1.75/2 = 0.875 \text{ in.} \quad \text{Inside radius of center post}$$

$$A_p = \pi(R^2 - R_i^2) = 32.08 \text{ in}^2 \quad \text{Cross sectional area of center post}$$

$$r_p = 0.5\sqrt{R^2 + R_i^2} = 1.713 \text{ in}^2 \quad \text{Radius of gyration of center post}$$

$$\lambda = \frac{K_p L_p}{r_p} \frac{1}{\pi} \sqrt{\frac{S_Y}{E}} = 0.888 \quad \text{From F-1334.3 (ASME 1998).}$$

Since the center post is a heavy shape, F-1334.3(b)(2) applies.

For $\lambda < 1$

$$P_Y = S_Y(A_p) = 63.52 \times 10^4 \text{ lb}$$

$$P = \frac{1 - \frac{\lambda^2}{4}}{1.11 + 0.75\lambda + 0.83\lambda^2 - 0.81\lambda^3} P_Y = 53.64 \times 10^4 \text{ lb}$$

The axial buckling load on the center post of the lowest of the six baskets is

$$P_{cp} = 5(35)(2,400) = 42.0 \times 10^4 \text{ lb}$$

and the ratio of load to capacity is

$$\text{Ratio} = \frac{P_{cp}}{P} = 0.783$$

Thus, the bottom basket center post capacity is adequate to support the entire 35 g vertical drop load. Because of deflection of the bottom plate under load, the load is shared between the center post and the support rods, and the center post does not carry the entire load. Therefore, buckling of the center post is not of concern.

For the horizontal drop, the center post is loaded in beam bending. Since the post is attached essentially rigidly to the bottom plate and is simply supported at the top coupling, it may be modeled as a propped cantilever. The loading consists of a uniform loading of eight fuel

elements (60° section), plus the center post weight, using 101 g. The total weight of the center post is bounded by 220 lb, and the length of the center post is bounded by 22 in. Each fuel element weighs 39.7 lb. The horizontal loading (1 g) is therefore

$$W_T = \frac{220 + 8(39.7)}{22} = 24.4 \text{ lb/in.}$$

The moment in a propped cantilever is maximum at the wall (i.e., at the joint to the bottom plate), which is the location of the threaded joint and of the minimum moment-resisting cross section. The maximum moment is

$$M_P = \frac{W_T L_P^2}{8} (101) = 149,096 \text{ in-lb.}$$

The center post is attached to the bottom plate using a single stub acme thread having a 5-in. diameter and 2-pitch. From the Machinery Handbook (Green and McCauley 1996), the stress area, A_t , and thread shear area, A_s , of the threaded portion are

$$A_t = \pi \left(\frac{E_s + K_s}{4} \right)^2 = 17.507 \text{ in}^2$$

$$A_s = \pi K_n \left[0.5 + \frac{1}{p} \tan(14.5^\circ)(E_s - K_n) \right] = 7.977 \text{ in}^2/\text{in.}$$

where

$$E_s = 4.7973 \text{ in.}$$

$$K_s = 4.6454 \text{ in.}$$

$$K_n = 4.7250 \text{ in.}$$

Although the boss thickness of the bottom plate is 1.5 in., the minimum thread engaged length is 0.922 in. because of the thread relief groove. Conservatively, 0.9 in. is assumed in analysis.

The shear stress in the threads from the applied moment at the joint, M_P , is found as follows. The moment is opposed by a couple having a force F along the post axis, with a pivot point a distance, R , of 3.313 in. away at the outer diameter of the post. The force is

$$F = \frac{M_P}{R} = 45,003 \text{ lb.}$$

Conservatively assuming that only the threads on the side of the post far from the pivot are active in resisting the load (i.e., half of the thread area), the shear area is

$$A_{sa} = (0.9)(0.5)A_s = 3.590 \text{ in}^2$$

where the factor of 0.9 is the effective length of the threads, and the factor of 0.5 accounts for only a half-circumference of thread. The shear stress in the threads is

$$\tau = \frac{F}{A_{sa}} = 12,536 \frac{\text{lb}}{\text{in}^2}$$

The allowable average primary shear stress for the drop condition is 26.3 ksi. The stress ratio on center post attachment thread shear is

$$\text{Ratio} = \frac{\tau}{26,300} = 0.477$$

The threaded portion also is subject to bending stress on the cross section. The moment of inertia of the threaded portion is

$$I = \frac{\pi}{64}(d_{\text{eff}}^4 - d_b^4) = 19.33 \text{ in}^4$$

where the minimum diameter of the thread relief, d_{eff} , is 4.60 in., and the bushing bore diameter, d_b , is 2.71 in. The c-distance is d_{eff} divided by two, or 2.3 in. The bending stress is

$$\sigma = \frac{M_p c}{I} = 17,740 \frac{\text{lb}}{\text{in}^2}$$

The allowable stress for primary membrane plus bending is 60.1 ksi (elastic analysis). The stress ratio is

$$\text{Ratio} = \frac{\sigma}{60,100} = 0.295$$

Therefore, the center post and its attachment to the bottom plate are acceptable.

Center Coupling. The center coupling is an extension of the center post that interfaces with adjacent baskets and the MCO shield plug. It is loaded during lifting and during a horizontal drop. During deadweight stacking, the center coupling interface with the center bushing is dimensioned to prevent loading of the center coupling. The minimum section of the coupling is the lifting grapple interface, which has an outer diameter of 2.66 in. and an inner diameter of 2.00 in. The net section was checked for tension (lifting load) and shear (horizontal drop loading):

$$A_{\text{MIN}} = [(2.66)^2 - (2.00)^2] \frac{\pi}{4} = 2.416 \text{ in}^2$$

$$\sigma_{\text{Lift}} = \frac{2,400}{A_{\text{MIN}}} = 993 \frac{\text{lb}}{\text{in}^2}$$

$$\text{Ratio} = \frac{\sigma_{\text{LIFT}}}{\frac{S_Y}{3}} = 0.142$$

Thus, the center coupling is adequate for the lifting load condition.

For the horizontal drop, the minimum section was checked for average shear adequacy. Conservatively ignoring bending resistance of bottom plate connection (top end reaction = $\frac{1}{2}$ of 101 g loading of eight 39.7-lb fuel pins)

$$\tau_{\text{DROP}} = \frac{8(39.7\text{lb})(101 \text{ g})}{2(A_{\text{MIN}})} = 6.64 \times 10^3 \frac{\text{lb}}{\text{in}^2}$$

$$S_U = 62,500 \text{ lb/in}^2$$

$$\text{Ratio} = \frac{\tau_{\text{DROP}}}{0.42 S_U} = 0.253$$

Therefore, the center coupling also is adequate for the horizontal drop 101 g loading.

Support Rod. The axial load carrying capacity of the support rods is determined using the approach outlined in the ASME Boiler and Pressure Vessel Code, Appendix F, Subsection F-1334.3(a)(1) (ASME 1998). According to this methodology, the allowable capacity

of the support rods is two-thirds of the maximum capacity determined by a nonlinear buckling finite element analysis, taking account of material plasticity and load eccentricity. The support rod is 21.967 in. long and is bolted securely to the baseplate. The material is type 304L stainless steel, certified as having the mechanical properties of type 304.

The ANSYS 5.4 finite element analysis model of the rod is shown in Figure 3-8. The rod is conservatively given a length of 22 in. Part of the shell is included to model the lateral support provided by the shell as the support rod's deformed shape moves outward. The rod is constructed of SOLID45 elements, and the MCO shell of SHELL43 elements, both capable of large deflections and nonlinear material behavior. The potential contact between the rod and the shell is modeled using CONTACT52 gap elements. Conservatively, the maximum gap (considering fabrication tolerances) of 0.265 in. is used,

Since there are six support rods, a one-sixth (60°) symmetry model was used. The bottom of the support rod is considered fixed to the bottom plate of the lowest basket. The MCO shell is 0.5 in. thick and is joined to the bottom plate of the canister at essentially the same elevation as the lower end of the support rod. The upper end of the shell segment is located 22 in. (approximately one basket length) above the top of the support rod. Since the support rod forces are very localized, this length is adequate. The circumferential edges of the shell are constrained in a manner consistent with symmetry, the bottom of the shell is fixed, and the top is free.

The load is applied to the support rod as a defined displacement in the axial direction and is applied at a node located 0.25 in. radially inward from the centroid of the rod. This offset provides the initial buckling eccentricity and is derived from the chamfer located on the lower surface of the bottom plate. Based on the radius of the chamfer (11.063 in.) and the radial location of the rod (outer edge at a radius of 11.226 in.), the radial distance of the top face of the rod that is not loaded by the bottom plate is $(11.226 - 11.063) = 0.163$ in. However, a load offset equal to the entire 0.25-in. chamfer is conservatively used. This radially inward bias of the load also ensures that the initial buckling of the rod will be in the outward direction, resulting in lateral support from the MCO shell. In addition, the radial displacement of the loaded node at the upper end of the rod is constrained to be zero, consistent with a pinned end. This requires that adequate friction be present between the bottom plate of the basket above and the support rod. As is demonstrated, the resulting radial reaction load is easily obtained with a conservative, lower bound friction coefficient. The design agent determined that the lowest dynamic, dry coefficient of friction for steel on steel is 0.42. However, a value of 0.1 may be conservatively assumed, which is adequate to pin the upper end of the support rod. The material properties of the support rod and MCO shell are based on a bilinear stress strain curve for type 304 material, using the yield strength of 23,250 lb/in² at 270 °F and a tangent modulus of 160,000 lb/in².

The resulting force-deflection curve for a single support rod is shown in Figure 3-9. The choice of 0.25 in. for the displacement of the top of the rod is arbitrary and ensures that the maximum capacity of the rod is developed. The first, lower peak is the initial bifurcation point as the rod begins to buckle. The lowest point is reached just before the rod first contacts the MCO shell, at which point the buckling capacity increases steeply. The curve levels off as an inward deformation of the lower part of the rod begins to occur. The stress in the shell remains elastic

throughout. The maximum load-carrying capacity of a single support rod is 65,625 lb at an axial deformation of 0.2 in. The radial reaction of the top of the rod is 6,400 lb, and to keep it in place, a minimum coefficient of friction of $6,400/65,625 = 0.098$ is needed, which is below the lower bound coefficient discussed above. Therefore, the pinned assumption of the upper end of the rod is valid. The deformed shape of the rod at an upper displacement of 0.2 in. is shown in Figure 3-10.

Since there are six support rods per basket, the total load carrying capacity of the support rods in the lowest basket is $6 \times 65,625 = 393,750$ lb. The length of the rods may vary by 0.030 in., but since the force-deflection curve is relatively flat in the region of maximum capacity, the effect of length variation of this modest magnitude is negligible. According to the ASME Boiler and Pressure Vessel Code, Appendix F, Subsection 1334.3(a)(1) (ASME 1998), the allowable capacity is two-thirds of this value, or

$$P_a = (2/3)(393,750) = 262,500 \text{ lb}$$

The load applied to the support rods is based on the load distribution between the center post and the rods and is a function of the stiffness of the bottom plate. The Figure 3-11 ANSYS model was used to determine the load distribution. The fuel pressure was applied to the plate, and the support rod and center post reactions were 5,209 lb and 3,778 lb, respectively, for a total load of 8,987 lb. The load-sharing ratios are therefore $3,778/8,987 = 0.42$ for the center post, and $5,209/8,987 = 0.58$ for the support rods. This analysis assumed simultaneous contact at all six rods and the center post. As stated above, the length of the rods and the center post could individually vary by 0.030 in. However, since the support rods retain essentially their maximum capacity up to and beyond a displacement of 0.2 in., the height variation of 0.03 in. has no effect on the load distribution. The load the support rods must carry is

$$P = (5)(2,400)(35)(0.58) = 243,600 \text{ lb}$$

which is equivalent to 58% of the weight of five Mark IA storage baskets weighing 2,400 lb each under a 35 g impact. The stress ratio is

$$\text{Ratio} = \frac{P}{P_a} = 0.928$$

Note that at the maximum applied load of 243,600 lb (40,600 lb/rod), the axial displacement of each rod is only 0.014 in., and thus, the rods have still not reached the first, lower bifurcation point. Therefore, the margin of safety is substantial. Two other conservatisms in the analysis are worthy of note. First, the relatively large flat surface on the top of the support rod leads to a degree of load centering, as shown in Figure 3-12. The rotation of the top of the support rod under eccentric loading has the effect of moving the load application point outward, thus reducing the eccentricity. In the model, by contrast, the eccentricity was not only upper bounded, but held

constant. Second, Paragraph F-1322.3© of the ASME Boiler and Pressure Vessel Code (ASME 1998) permits an adjustment of the stress-strain properties to account for strain rate effects. The increase in the yield strength for the drop accident load cases was conservatively ignored. Therefore, buckling of the support rods in an end drop is not of concern.

The support rods are attached to the base plate with 5/8 -11 flat head screws. Torque is applied to the limits of the ASME Boiler and Pressure Vessel Code, Subsection NG-3232.2 (ASME 1998), the preload stress being limited to $(1.2)0.9S_Y$. For SA-193 material at 270 °F

$$S_Y = 3(S_M) = 3(11,600) = 34,800 \text{ lb/in}^2.$$

The stress limit is therefore

$$(1.2)(0.9)(34,800) = 37,584 \text{ lb/in}^2.$$

The maximum allowed preload force is therefore

$$F = \sigma A_s = 8,494 \text{ lb}.$$

The stripping of the thread in the support rod was checked for this loading:

$$\frac{A_{S,INTERNAL}}{L_E} = \pi n D_{SMIN} \left[\frac{1}{2n} + 0.57735 (D_{SMIN} - E_{NMAX}) \right]$$

where

D_{SMIN} = minimum major diameter of external thread = 0.6113 in.

E_{NMAX} = maximum pitch diameter of internal thread = 0.5732 in.

L_E = length of engagement
 = 2.50 (length of bolt) - 1.25 (maximum thickness of base plate)
 = 1.25 in.

$$\frac{A_{S,INTERNAL}}{L_E} = \pi (11)(0.6113) \left[\frac{1}{2(11)} + 0.57735 (0.6113 - 0.5732) \right] = 1.425 \frac{\text{in}^2}{\text{in}}$$

$$A_{S,INTERNAL} = 1.425 (1.25) = 1.781 \text{ in}^2.$$

The allowable stress is

$$0.6 S_M = 0.6(16,700) = 10,020 \text{ lb/in}^2.$$

The stripping load is

$$(1.781)(10,020) = 17,848 \text{ lb} > 8,494 \text{ lb preload.}$$

Therefore stripping of the threads is not a concern.

The thread friction coefficient, k , is 0.2. The calculated torque with $d = 0.625$ in. is

$$T = Fkd/12 = 88 \text{ lb-ft.}$$

With an uncertainty of 30%, the torque should be limited to 68 lb-ft. The recommended torque is 60 ± 8 lb-ft.

The use of a clamped condition for the attachment of the support rod to the base plate is substantiated by investigating the worst-case bending moment occurring during buckling to determine the bolt load induced by the resulting moment around the attachment interface of the rod to base plate. This load is then compared with the colinear axial force acting at the same time (see Figure 3-13. Using the parameters provided in Table 3-23

$$Y = \frac{\Sigma AY}{\Sigma A} = \frac{0.2313}{0.6952} = 0.3327 \text{ in}^2$$

$$I = \Sigma I_O + \Sigma AY^2 - \Sigma A(Y)^2 = 0.0083 + 0.1141 - (0.6952)(0.3327)^2 = 0.0454 \text{ in}^4$$

For the worst-case moment point (at the end of the analysis, maximum deflection, where the rod has the greatest curvature, and the axial load is starting to fall), the fixed end moment is 14,922 in-lb.

The bolt load is

$$P_{\text{BOLT}} = Mc/I (A_B) = (14,922)(0.682 - 0.3327)/(0.0454)(0.226) = 25,918 \text{ lb}$$

where c is the distance from bolt centerline to neutral axis.

The colinear axial force directly downward on the base plate at this time step is 64,698 lb. Since this is much more than the bolt force, it is clear that the bolt is not necessary to keep the base of the rod "fixed." The axial buckling load of 64,698 lb against the flat bottom is more than enough to do that. A check of the axial reaction loads of all 40 of the nodes on the bottom face of the rod shows all except three (located on the radial outside of the rod) are pressing down onto the base plate. Therefore, the rod bolt is locational only, and no stress analysis in connection with the buckling analysis is required.

Table 3-23. Support Rod Geometric Properties.

Section	Area	Y	AY	AY ²	I ₀
1	0.4027	0.1664	0.0670	0.0111	0.0037
2	0.0762	0.2218	0.0169	0.0037	0.0005
3	-0.0097	0.3196	-0.0031	-0.0010	neg
4	0.226	0.6820	0.1541	0.1051	0.0041
$\Sigma \rightarrow$	0.6952	--	0.2313	0.1141	0.0083

Bottom Plate. The Mark IA bottom plate was evaluated for normal operation and drop accident conditions. As indicated in the following paragraphs, the bottom plate design is controlled by the vertical drop load event.

As indicated in Appendix A of the MCO Performance Specification (Goldmann 2000b), a loaded MCO consists of six Mark IA baskets. The basket bottom plate stresses would be relatively low if the basket support rods were all aligned. However, since the baskets are not indexed to ensure support rod alignment, rotational offsets are expected. The maximum bottom plate bending occurs when the support rods are midway between the above basket rods (30° offset). The critical bottom plate is the next-to-the-bottom basket with the bottom basket rotated 30° from the basket above. For this condition, the critical bottom plate rods react to the loading from the top four baskets. This support rod offset produces significant bending stresses in the bottom plate, as demonstrated below.

The bounding drop loads for the bottom plate are a vertical drop of 35 g and a horizontal drop of 101 g. As indicated below, the critical drop for the bottom plate is the 35 g vertical drop. Since the drop load stresses are much greater than the normal operation stresses within the MCO, the drop load condition controls the bottom plate design. However, since the boundary conditions and acceptance criteria of the single basket lift condition are much different than those for the drop condition, the lifting condition also is addressed to confirm adequacy for this load condition.

Two ANSYS models were developed for evaluating the bottom plates of the Mark IA basket. The first model generated was a 60° sector model complete with holes, as shown in Figure 3-11. The mesh refinement necessary to properly define the holes resulted in a relatively large number of elements that ran relatively slowly. The second model generated was a 180° model without holes (Figure 3-14) and with three support rods, one every 60°. Note that in both

models, shell elements (SHELL43 and SHELL63) were used to model the center post⁶ and bottom plate, while solid elements (SOLID45) were used to model the support rod.

Symmetry boundary conditions (no rotation about a radius line nor displacement in the tangential direction) were used at the zero and 60° edges of the first model. For the stacking load condition within the MCO, it was assumed that the supporting basket below was rotated 30°, resulting in the support rods beneath being centered on the edges of the model. The precise bottom-plate-to-support-rod contact locations were not known. To address this contact issue, gap elements were placed at the bottom plate-support rod interface.

A 180° degree symmetry model, shown in Figure 3-14, was developed for the horizontal drop evaluation. The primary purpose of this model was to demonstrate conformance to the 2-in. distortion limit on the center post (Goldmann 2000b, Section 4.9.3). The bottom plate and center post were modeled using ANSYS SHELL43 elements. The support rods were modeled with beam elements (BEAM4). Note that beam elements also were used at the support rod-bottom plate connection locations to spread the support rod moments over the outlined area of the rod. Although structural adequacy of the support rods is not an issue for the horizontal drop, the support rods were modeled to introduce the support rod inertia loads (moments) to the bottom plate. Gap elements (CONTAC52) were used at the drop-side interface with the MCO in order to achieve a reasonable interface load distribution. Since the horizontal drop can be preceded by a vertical drop, rigid links (BEAM4) were used to account for a potential axial reaction offset in the bottom plate occurring during the vertical drop (an offset of 0.5 in. was used).

Using a no-hole model, an initial estimate of the 35 g vertical drop stresses was generated. The loading and boundary conditions corresponded to a next-to-the-bottom basket, with four Mark IA baskets above. Before the vertical drop was analyzed, a run was made to estimate the fuel load distribution between the center post and support rods by applying a uniform pressure to the bottom plate to simulate the fuel inertia loading that was reacted through the center post and support rod. As discussed in the support rod section, under elastic conditions, 58% of the fuel loading is reacted through the support rods, and 42% of the fuel loading is reacted through the center post. This support rod-center post load distribution provides a reasonable, and likely conservative, estimate of the support rod loading in the elastic vertical drop analysis described below.

⁶The model was built assuming a center post made from 6-in., XXS pipe. The actual center post is made from solid bar with a 1.75-in.-diameter hole. In the vertical drop analysis, the center post plays only a negligible role because the stresses in the bottom plate do not depend on the center post configuration. In the horizontal drop analysis, the use of a pipe in the finite element model is conservative. This is because the actual center post, since it is much stiffer than the pipe, will impart a smaller out-of-plane rotation to the bottom plate, thus reducing the stress in the bottom plate. For these reasons, the use of a center pipe configuration in the models is conservative.

Subsequent Mark IA basket design changes have increased the basket weight. The change from a center pipe to a center post added approximately 80 lb more, but all of the added weight is carried directly by the relatively stiff center post itself, and does not affect stresses in the bottom plate. Therefore, although the weight has been specified as bounded by 2,400 lb, a weight of 2,300 lb can be used for this analysis without significant loss of accuracy. Using this weight and 35 g, the four basket loading is

$$(35)(2,300)(4) = 322,000 \text{ lb}$$

For an elastic drop analysis, it is reasonable to assume that the support rods equally share the support rod portion (58%) of the 322,000 drop load, resulting in a support rod loading for the next-to-the-bottom basket of

$$F_r = (0.58)(322,000)/6 = 31,127 \text{ lb}$$

The center post load estimate for a 60° sector is

$$F_{cp} = (0.42)(322,000)/6 = 22,540 \text{ lb}$$

These loads were imposed on the model in Figure 3-8 in the form of pressures on top of the support rod (10,831 lb/in²) and center post (8,647 lb/in²). Vertical constraints were introduced at the bottom of the center post and at the interface with the bottom basket support rods. Fuel inertia loading on the bottom plate was input as an equivalent pressure, using 39.7 lb/fuel rod (Goldmann 2000b, Appendix A):

$$F_f = 48(39.7)(35) = 66,696 \text{ lb}$$

$$A_p = [(22.625)^2 - (6.625)^2 - 102(0.51)^2](\pi/4) = 346.93 \text{ in}^2$$

$$A_r = 2.874(6) = 17.24 \text{ in}^2$$

$$\text{Pressure} = F_f/(A_p - A_r) = 202.30 \text{ lb/in}^2 \text{ (Equivalent fuel pressure, plate with 102 0.5-in. holes).}$$

The elastic analysis results are summarized in the Figure 3-15 stress intensity contour plots. Note that for the elastic analysis, the peak stress intensities are adjacent to the flow holes. These peak stresses are considered very localized, and the maximum is 68.6 ksi. Comparing the maximum plate bending stress of 68.6 ksi with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Level D allowable stress intensity of 60.12 ksi ($1.5 \times 2.4S_M$ at 132 °C) indicates a small overstress for the elastic analysis limits. This elastic overstress was resolved by performing the elastic/plastic analysis described below.

Using the Figure 3-11 model, an elastic/plastic analysis of the 35 g drop was performed, assuming bilinear plasticity. The method for developing the bilinear stress-strain curve was obtained from Pugh (1972), adapted from 304 stainless steel data. Pugh (1972) indicates the strain hardening coefficient is relatively independent of temperature. A value of $0.16 \times 10^6 \text{ lb/in}^2$ was obtained from Table B.1 of Pugh (1972), for a conservative maximum strain of 5%. Since

304 stainless steel and 304L stainless steel are nearly identical materials, it is reasonable to use the same strain hardening coefficient and the 17.8% yield increase.

Maximum stress intensities of 71.5 ksi (top fiber) and 78.5 ksi (bottom fiber) at the hole adjacent to the support rod were reported (see Figure 3-16). The ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Paragraph NG-3213.10 (ASME 1998), classifies a peak stress as a stress "...that does not cause any noticeable distortion...", is "...at a local structural discontinuity...", and is "...highly localized..." The stresses reported (71.5 ksi and 78.5 ksi) are at a local discontinuity (flow hole), highly localized, and do not cause any noticeable distortion. Node 495 reports a displacement of 0.0202 in. (node 495 is at the "base" of the localized stress). Nodes 3318, 5622, and 9232, away from the discontinuity, have reported displacements of 0.0183 in., 0.0184 in., and 0.0196 in., respectively. Therefore, since the displacements listed are very similar, the reported stresses (71.5 ksi and 78.5 ksi) are classified as peak stresses. Results listed in Table 3-24 for the top and bottom fiber of the bottom plate reflect stresses away from the discontinuity (away means about 1 or 2 radius lengths from the hole).

Table 3-24. Vertical Drop Stress Intensity Results Summary.

Component		Stress category	Stress intensities (ksi)		Ratio
			Maximum	Allowable	
Bottom plate	Middle	$P_m (0.7S_u)$	14.1	43.8	0.32
	Top	$P_m + P_b (0.9S_u)$	33.4	56.3	0.59
	Bottom	$P_m + P_b (0.9S_u)$	36.7	56.3	0.65
Center post	Middle	$P_m (0.7S_u)$	13.5	43.8	0.31
	Top	$P_m + P_b (0.9S_u)$	30.5	56.3	0.54
	Bottom	$P_m + P_b (0.9S_u)$	16.1	56.3	0.29

P_m = primary membrane stress.

P_b = bending stress.

The vertical drop stress intensity contours are shown in Figure 3-15. A tabular summary of the membrane and membrane plus bending results are provided in Table 3-24. Note that all predicted stress intensity ratios are less than one, indicating that the results are within ASME Boiler and Pressure Vessel Code allowables.

The plastic strain contours are shown in Figure 3-17. Note that a maximum plastic strain of 0.1% was predicted, which is well below the 5% assumed for the strain hardening coefficient selection above. Also, note that a maximum displacement of 0.033 in. was predicted. This maximum displacement occurs in the vertical direction and occurs in the plate below the support rod. The bottom plate distortion is of interest because of the potential for a horizontal drop

following a vertical drop. As indicated in the bottom plate section, this maximum plate distortion was considered in the horizontal drop evaluation.

Using the 180° model shown in Figure 3-14, an elastic/plastic analysis was performed for the 101 g horizontal drop load condition. The plastic analysis option was selected for two reasons: (1) an initial elastic analysis predicted local stresses that exceeded the elastic allowables, and (2) total transverse distortion (elastic plus plastic) predictions were needed to demonstrate conformance to the MCO Performance Specification (Goldmann 2000b, Section 4.19.3) nuclear criticality safety requirement that the "void space centerline shall not deviate more than two inches from the MCO centerline."

As indicated by the small elements on the right side (impact side) of the model shown in Figure 3-14, line elements were used to account for the potential offset associated with bottom plate plastic distortions occurring during a preceding vertical drop. Note from Figure 3-18 that offset from the bottom plate centerline can also be affected by the bottom plate edge contact. Although the vertical drop distortion was less than 0.1 in., an offset of 0.5 in. was conservatively assumed. Although the actual offset would be limited to a small area (near a support rod), a uniform offset was conservatively assumed.

The acceleration loading was increased to 1.5 times the specified drop loading of 101 g to ensure that the ASME Boiler and Pressure Vessel Code, Level D buckling requirements were met (ASME 1998, Paragraph F-1331.5(a), loading less than two-thirds buckling capacity). An earlier elastic buckling analysis, with no bottom plate offset, indicated that the elastic buckling strength was in excess of 1,000 g (stress > yield), which indicates that the actual buckling failure mode is inelastic.

The nonlinear results for the horizontal drop evaluation are summarized in Figure 3-19. Note in Figure 3-19 that a maximum stress intensity of 32.1 ksi was predicted for the top side of the bottom plate. The corresponding maximums for the shell middle and bottom surfaces are 18.6 and 31.1 ksi, respectively. Since the maximum surface stress intensity is more than 50% higher than the midsurface results, membrane plus bending stress intensity is the critical value for the horizontal drop. The allowable membrane plus bending stress intensity (plastic analysis) is 56.3 ksi, resulting in a ratio of

$$\text{Ratio} = 32.1/56.3 = 0.570 .$$

As indicated, a loading 1.5 times the 101 g drop loading (151.5 g) does not result in an unstable response. Thus, the ASME Boiler and Pressure Vessel Code, Level D requirement (load less than two-thirds buckling strength) is met.

A criticality control limit in Section 4.19.3 of the Performance Specification (Goldmann 2000b), specifies that the center post cannot deviate from the MCO centerline by more than 2.00 in. From Figure 3-19, a maximum horizontal drop displacement of 0.51 in. is indicated in the figure legend (101 g loading). This maximum displacement occurs at the top of a support rod, which has no criticality concern. From Figure 3-19, the maximum displacement in

the bottom plate is only 0.08 in. for a 101 g loading. Combining this value with the 0.1875-in. radial displacement caused by the difference between the basket outer diameter and the MCO inner diameter results in a deviation of 0.268 in. The resulting allowable ratio is

$$\text{Ratio} = 0.268/2.00 = 0.134 .$$

Scrap Basket Shroud. The scrap basket shroud extends the full height of the basket and is used to contain the scrap pieces. The scrap pieces vary in size and shape, and resulting pressure varies significantly. A reasonable estimate of the scrap pressure can be obtained by considering wall pressures associated with angular rock (e.g., limestone, iron ore). The scrap pressure during drop loading was not considered because it is assumed that the drop loading scrap pressure is carried by the MCO wall. An evaluation of the wall pressure and corresponding structural evaluation of the shroud was performed, which revealed the maximum membrane pressure stress in the shroud (lifting load condition).

$$P_m = 155 \text{ lb/in}^2$$

$$\text{Ratio} = \frac{P_m}{\frac{S_y}{3}} = 0.061 .$$

Therefore all hoop connections (welding, brazing, or mechanical) performed on the shroud must have a minimum strength equivalent to that of the calculated maximum membrane pressure stress calculated above.

The copper subassembly of the scrap basket is designed to withstand a distributed load in tension on the outside shroud of 10,350 lb before yielding and 17,250 lb before failure. This provides a safety factor of three to yield and five to failure during loading of the basket into the MCO.

The maximum allowable stress is 5.2 ksi at 250 °F and 5.1 ksi at 300 °F (ASME 1998). Using linear interpolation, the maximum allowable stress at 270 °F is 5.13 ksi. Using this in relation to the allowable stress of 6.70 ksi at room temperature, with the yield criteria governing, the yield stress at room temperature is 10 ksi. Therefore, the yield stress at 270 °F is as follows:

$$S_y = \frac{5.13}{6.70}(10) = 7.65 \text{ ksi} .$$

The screw specified is a self-tapping flat head, number 10, Type AB (square drive, 18-8 stainless steel material). The minimum head diameter is 0.389 in. at the end of the head and

0.172 in. at the inside of the shroud (i.e., at 0.125 from the flat surface of the screw head). The average head diameter bearing in the copper shroud is calculated as follows:

$$\text{Average head diameter} = \frac{0.389 + 0.172}{2} = 0.281 \text{ in.}$$

Limiting the bearing stress in the copper to yield, the allowable load when yielding occurs is calculated:

$$P_{\text{BEARING}} = (F_Y)(\text{Area}_{\text{BEARING}}) = (7,650)(0.281)(0.125) = 269 \text{ lb/screw.}$$

Checking the shear-out failure along sections at 40° from the loading direction results in

$$P_{\text{SHEAR-OUT}} = (\text{shear allowable stress})(2T)[\text{edge margin} - (D/2)\cos 40^\circ]$$

at an edge margin of 0.50 in.

$$\begin{aligned} P_{\text{SHEAR-OUT}} &= (0.6)(7,650)(2)(0.125)[0.5 - (0.389/2)0.766] \\ &= 403 \text{ lb/screw at 0.50 in. edge margin.} \end{aligned}$$

Therefore, shear-out will not occur.

The shear failure of the screw also was checked. The diameter of the screw is 0.164/0.157 in. The minimum allowable stress for 18-8 bolting according to the ASME Boiler and Pressure Vessel Code (ASME 1998) is 7.74 ksi at 270 °F.

$$\begin{aligned} \text{Shear allowable of screw} &= 0.6(2S_M)(\text{shear area}) \\ &= 0.6(2)(7,740)(\pi/4)(0.157)^2 = 180 \text{ lb/screw.} \end{aligned}$$

Therefore, the shear stress in the screw governs the strength of the attachment screws. With 10 screws per segment, the stress ratio is calculated as follows:

$$\text{Ratio} = \frac{10,350/6}{10(180)} = 0.96$$

Thermal Expansion. As stated above, the shroud, divider plates and fines divider tube are fabricated out of copper, and the bottom plate is fabricated out of type 304L stainless steel. Since copper has a higher coefficient of thermal expansion than stainless steel, a thermal expansion analysis was performed. The evaluation was performed for a temperature difference of 200 °F (going from 70 °F to 270 °F). This thermal expansion analysis was provided for informational and operational reasons because the copper components are nonstructural items.

Vertical expansion for the stainless steel plate was calculated as follows:

$\alpha_{ss} = 8.94 \times 10^{-6} \text{ in/in/}^{\circ}\text{F}$	Stainless steel coefficient of thermal expansion
$\Delta T = 200 \text{ }^{\circ}\text{F}$	Temperature differential
$L_{ss} = 1.25 \text{ in.}$	Bottom plate height
$\Delta L_{ss} = \alpha_{ss} L_{ss} \Delta T = 0.002 \text{ in.}$	Vertical expansion of bottom plate.

Vertical expansion for the copper shroud was calculated as follows:

$\alpha_{cu} = 10.0 \times 10^{-6} \text{ in/in/}^{\circ}\text{F}$	Copper coefficient of thermal expansion
$\Delta T = 200 \text{ }^{\circ}\text{F}$	Temperature differential
$L_{cu} = 21.5 \text{ in.}$	Shroud height
$\Delta L_{cu} = \alpha_{cu} L_{cu} \Delta T = 0.043 \text{ in.}$	Vertical expansion of copper shroud.

Therefore, the total vertical thermal expansion for the stainless steel and the copper at the circumference of the basket is

$$\Delta L = \Delta L_{cu} + \Delta L_{ss} = 0.045 \text{ in.}$$

The vertical thermal growth of the stainless steel post is calculated as follows:

$L_{cp} = 23.2 \text{ in.}$
$\Delta L_{cp} = \alpha_{ss} L_{cp} \Delta T = 0.041 \text{ in.}$

One can conclude that there is no significant differential expansion between the center post and the shroud. Therefore, the 1-in. vertical gap left for the center post expansion is adequate to ensure no interference fit between the shroud and the bottom plates of the adjacent baskets.

Radial expansion for the stainless steel plate is calculated as follows:

$\alpha_{ss} = 8.94 \times 10^{-6} \text{ in/in/}^{\circ}\text{F}$	Stainless steel coefficient of thermal expansion
$\Delta T = 200 \text{ }^{\circ}\text{F}$	Temperature differential
$R_{plate} = 11.31 \text{ in.}$	Bottom plate outside radius
$\Delta L_{ss} = \alpha_{ss} D_{plate} \Delta T = 0.020 \text{ in.}$	Radial expansion of bottom plate.

Radial expansion for the copper shroud is calculated as follows:

$\alpha_{cu} = 10.0 \times 10^{-6} \text{ in/in/}^\circ\text{F}$	Copper coefficient of thermal expansion
$\Delta T = 200^\circ\text{F}$	Temperature differential
$R_{shroud} = 11.31 \text{ in.}$	Shroud outside radius
$\Delta L_{cu} = \alpha_{cu} R_{shroud} \Delta T = 0.023 \text{ in.}$	Radial expansion of copper shroud.

The copper shroud is expected to expand radially 0.023 in. while the stainless steel MCO shell and basket bottom plate are expected to expand by approximately 0.020 in. As these values are very comparable, the fabrication gaps will remain open and no undetermined loads will be applied to the MCO shell from unexpected basket expansion loads.

Summary. A summary of the component stress analysis results is provided in Table 3-25. Note that the predicted maximums are below allowables for all components and conditions.

3.2.3.1.5 Basket–Multi-Canister Overpack Interface Components. An important issue in the horizontal drop evaluation in Section 3.2.3.1.4 is the assumption that the top end of each Mark IA basket (center post) remains inside (i.e. overlaps) the mating component during the drop. This assumption is vital to the horizontal drop calculations because the finite element analysis assumed a transverse support for the top end of the center post. Also, Section 4.19.3 of the Performance Specification (Goldmann 2000b) (criticality control) requires that the center post not deviate by more than 2 in. from the MCO centerline. This section provides justification for the assumption that the Mark IA baskets will not come apart at the interfaces.

The basket–MCO interfaces were designed to ensure that, during normal operations, the center coupling would maintain at least a 1-in. overlap with the shield plug assembly. If the MCO basket support plates were to collapse during a vertical drop, the center coupling overlap might be lost. Therefore, the basket support plates were evaluated to ensure that the ASME Boiler and Pressure Vessel Code, Level D axial compression limits (ASME 1998) were met for a vertical drop.

Table 3-25. Summary of Mark IA Storage Basket Stress Results.

Component	Critical load condition	Stress category	Predicted maximum	Allowable	Ratio
Center post-bottom plate threaded coupling	Lifting	Shear	334 lb/in ²	7,010(0.6) lb/in ²	0.079
Center post capacity	35 g vertical drop	Buckling	420,000 lb	536,400 lb	0.783
Center post-bottom plate threaded coupling	101 g horizontal drop	Shear	12,536 lb/in ²	26,300 lb/in ²	0.477
Center post-bottom plate threaded coupling	101 g horizontal drop	Pm + Pb	17,740 lb/in ²	60,100 lb/in ²	0.295
Center coupling	101 g horizontal drop	Shear	6,640 lb/in ²	26,250 lb/in ²	0.253
Center coupling	Lifting	Pm	993 lb/in ²	7,010 lb/in ²	0.142
Support rod	35 g vertical drop	Buckling	243,600 lb	262,500 lb	0.928
Bottom plate	35 g vertical drop	Pm	14,100 lb/in ²	43,800 lb/in ²	0.322
Bottom plate	101 g horizontal drop	Pm + Pb	32,100 lb/in ²	56,300 lb/in ²	0.570
Center post criticality control	Sequential 35 g vertical and 101 g horizontal drops	Plastic distortion and potential instability	0.268 in. radial displacement	2.00 in.	0.134
Support plate	Sequential 27 g vertical and 101 g horizontal drops	Buckling	13,180 lb/in ²	16,230 lb/in ²	0.812
Support plate weld	Sequential 27 g vertical and 101 g horizontal drops	Pm	12,450 lb/in ²	16,040 lb/in ²	0.776

Pm = primary membrane stress.

Pb = bending stress.

Assuming the following plate dimensions (six radial spokes at 60° intervals)

$t_p = 0.50$ in.	Support plate width
$l_p = 9.83$ in.	Support plate length
$h_p = 1.49$ in.	Support plate height

Per inch length of plate

$$I = \frac{(1)(t_p)^3}{12} = 0.01 \text{ in}^4$$

$$r = \sqrt{\frac{I}{(1)(t_p)}} = 0.144 \text{ in}$$

$$K = 2.1 \quad \text{Effective length, fixed/free}$$

$$S_Y = 19,800 \text{ lb/in}^2 \quad \text{304L yield stress at maximum drop temperature of } 132^\circ \text{C}$$

$$E = 27.2 \times 10^6 \text{ lb/in}^2 \quad \text{304L modulus at } 132^\circ \text{C}$$

The axial compression evaluation (ASME 1998, Paragraph F-1334.3) is as follows:

$$\lambda = \left(\frac{Kh_p}{r} \right) \left(\frac{1}{\pi} \right) \left(\sqrt{\frac{S_Y}{E}} \right) = 0.186$$

$0 < \lambda < 1$, then first equation applies

$$\sigma_{\text{ALLOW}} = (S_Y) \frac{1 - \frac{\lambda^2}{4}}{1.11 + 0.50\lambda + 0.17\lambda^2 - 0.28\lambda^3} = 16.23(10^3) \frac{\text{lb}}{\text{in}^2}$$

The maximum vertical acceleration level for a sequential drop is 27 g

$$W_b = 2,400 \text{ lb} \quad \text{Basket weight}$$

$$\sigma = \frac{6W_b(27)}{6t_p l_p} = 13.18(10^3) \frac{\text{lb}}{\text{in}^2}$$

$$\text{Ratio} = \frac{\sigma}{\sigma_{\text{ALLOW}}} = 0.812$$

Since the safety margin is small, it is noted that the results are conservative in that no credit is taken for either the vertical weld on the inside end of the plate or the lateral resistance for friction.

The support plate welds have been evaluated as follows:

$$F_p = \sigma t_p = 6.59 \times 10^3 \text{ lb/in}^2 \quad \text{Support plate drop loading/inch}$$

Assuming a maximum load offset of half the plate thickness

$$M_p = F_p \frac{t_p}{2} = 1.65(10^3) \text{ in-lb/in}$$

The 0.375-in. weld stress in the throat is calculated as follows:

$$\sigma_w = \frac{M_p}{t_p \frac{\sqrt{2}}{2} \frac{3}{8}} = 12.45(10^3) \frac{\text{lb}}{\text{in}^2}$$

The allowable weld stress is based on the base metal allowable:

$$2.4S_m = 40.1 \text{ ksi.}$$

From Table NG-3352 (ASME 1998), the weld quality factor for a double fillet weld is 0.4, so the weld allowable is as follows:

$$A_w = (40.1)(0.4) = 16.04 \text{ ksi.}$$

The stress ratio is calculated as follows:

$$\text{Ratio} = \frac{\sigma_w}{16,040} = 0.776$$

Thus, the basket support plates and attachment welds meet the ASME Boiler and Pressure Vessel Code, Level D compression load requirements (ASME 1998), and the overlapping (telescoping) of the baskets is not jeopardized.

At the top end of the basket stack, the basket interfaces with the bottom of the shield plug assembly (specifically, the basket stabilizer, shield ring, and shield plate [see Figure 1-1]). For a top end drop, the basket loading would be reacted by the 1-in.-thick shield ring (area = 67.5 in²) and the basket stabilizer. The shield ring area alone is more than double the bottom basket interface support plate area. Thus, the top axial support is adequate by comparison with the bottom basket interface.

For a horizontal drop, the top basket relies upon the shield plate for transverse support. For normal conditions, the shield plate position is maintained through a weld connection to the shield ring, which in turn is welded to the shield plug. If either or both of these welds should fail during a horizontal drop, the shield plate would be captured between the top basket and shield plug and would continue to support the top end of the basket. Therefore, the maximum center post movement would be the 0.25-in. difference between the inside radius of the MCO and the outside radius of the shield ring, which is well below the 2-in. limit in the MCO Performance Specification (Goldmann 2000b, Section 4.19.3).

3.2.3.2 Stress Analysis of the Mark IV Scrap Basket. The following sections summarize the design agent's evaluation of the Mark IV scrap basket for lifting and deadweight loads. The structural adequacy evaluation is based on the criteria of the ASME Boiler and Pressure Vessel Code (ASME 1998). A combination of hand calculations and ANSYS analysis have been used.

The primary components of the Mark IV scrap basket are listed in Table 3-26.

Table 3-26. Mark IV Scrap Basket Geometry.

Component	Description
Center pipe	Tubular section with outside diameter of 2.84 in. and a nominal wall thickness of 0.50 in. Pipe is attached to bottom plate with a 3.00-8 class 2 thread.
Bottom plate	The bottom plate is a 1.20-in.-thick plate with 0.5-in.-diameter holes.
Shroud	The shroud is 0.125-in.-thick copper. The outside diameter of the shroud is 22.625 in. and it is 26.40 in. high.
Support posts	Six support posts are located 60° apart around the periphery of the bottom plate. The posts are stainless steel and have a minimum diameter of 1.31 in.
Fines divider tube	The fines divider tube is rolled copper plate. It is 26.25 in. high and is 0.125 in. thick.
Shroud segments (stiffener plates)	Six copper radial stiffener plates connecting the fines divider tube to the shroud. The stiffener plates are 0.25 in. thick copper and 26.25 in. high.
Bottom screen	The bottom screen is stainless steel and is sandwiched between the bottom plate and the stiffener plates.
Shroud to bottom plate connection	The shroud is fastened to the bottom plate by means of a screw similar to the shroud attachment in the Mark IA scrap basket.

The Mark IV scrap basket structural components are fabricated from type 304L and type 304 stainless steel. Only type 304L material properties are used in the evaluation to preserve conservatism. The nonstructural shroud section assembly is fabricated from copper to enhance the heat transfer properties of the basket.

3.2.3.2.1 Lifting Loads. According to Section 4.12.3 of the MCO Performance Specification (Goldmann 2000b), the basket design "shall meet the safety factors of three on yield and five on ultimate material strength. The safety factors apply from 5 °C to 100 °C."

3.2.3.2.2 Deadweight Loads. According to Section 4.12.3 of the MCO Performance Specification (Goldmann 2000b), all baskets must be able to support the fuel at 1.0 g while at 132 °C (270 °F). The specification does not provide criteria for the Mark IV baskets under these loads, thus the normal condition (service level A) criteria of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME 1998) are used. As described in the following paragraphs, the criteria of Subsection NG are supplemented by the criteria of Subsection NF (ASME 1998) for the center pipe.

The Mark IV scrap baskets have been evaluated for two load cases:

1. **Lifting the basket and contents while at 100 °C** — This loading has been evaluated using criteria based on the safety factors incorporated in Section 4.12.3 of the MCO Performance Specification (Goldmann 2000b).
2. **Deadweight of the basket and contents while at 132 °C** — The basket inside the MCO is considered the limiting case. The basket may be at the top of the MCO (and thus be required to support only its own weight) or at the bottom of the MCO (where it is required to support the dead load of four additional baskets).

Section 4.12.2 of the MCO Performance Specification (Goldmann 2000b) exempts the Mark IV baskets from consideration of drop or other accident loads.

3.2.3.2.3 Stress Analysis Calculations. The Mark IV scrap basket is evaluated using the results of the evaluation of the Mark IV storage basket in Section 3.2.3.3.

Center Pipe, Support Rods, Bottom Plate, Center-to-Bottom-Plate Attachment. The structural portion of the Mark IV scrap basket is identical to the Mark IV storage basket. The weight of the Mark IV scrap basket is less than the weight of the Mark IV storage basket, therefore the stresses in the Mark IV scrap basket are bounded by the analysis results of the Mark IV storage basket as evaluated in Section 3.2.3.3, and no further analysis is required.

Basket Subassembly. Since the basket subassembly, shroud, and dividers are fabricated from copper (non-ASME Code material), and their connecting welds are not structural welds, no analysis is required. Welds shall be inspected using surface visual examination.

Shroud-to-Bottom-Plate Connection. The shroud is fastened to the bottom plate by screws in a pattern identical to that in the Mark IA scrap basket. This attachment was evaluated in the section on the Mark IA scrap basket, and no further analysis is required.

Thermal Expansion. As stated above, the shroud, divider plates, and fines divider tube are fabricated out of copper, and the bottom plate is fabricated out of type 304L stainless steel. Since copper has a higher coefficient of thermal expansion than stainless steel, a thermal expansion analysis was performed. The evaluation was performed for a temperature difference of 200 °F (going from 70 °F to 270 °F). The thermal analysis was provided for informational and operational reasons because the copper components are non-structural items.

Vertical expansion for the stainless steel plate was evaluated as follows:

$\alpha_{ss} = 9.70 \times 10^{-6} \text{ in/in/}^\circ\text{F}$	Stainless steel coefficient of thermal expansion
$\Delta T = 200 \text{ }^\circ\text{F}$	Temperature differential
$L_{ss} = 1.20 \text{ in.}$	Bottom plate height
$\Delta L_{ss} = \alpha_{ss} L_{ss} \Delta T = 0.002 \text{ in.}$	Vertical expansion of bottom plate.

Vertical expansion for the copper shroud was evaluated as follows:

$\alpha_{cu} = 10.0 \times 10^{-6} \text{ in/in/}^\circ\text{F}$	Copper coefficient of thermal expansion
$\Delta T = 200 \text{ }^\circ\text{F}$	Temperature differential
$L_{cu} = 26.4 \text{ in.}$	Shroud height
$\Delta L_{cu} = \alpha_{cu} L_{cu} \Delta T = 0.053 \text{ in.}$	Vertical expansion of copper shroud.

Therefore, the total vertical thermal expansion for the stainless steel plate and copper shroud at the circumference of the basket is calculated as follows:

$$\Delta L = \Delta L_{cu} + \Delta L_{ss} = 0.055 \text{ in.}$$

The vertical thermal growth of the stainless steel post was evaluated as follows:

$L_{cp} = 27.8 \text{ in.}$
$\Delta L_{cp} = \alpha_{ss} L_{cp} \Delta T = 0.054 \text{ in.}$

One can conclude that there is no differential expansion between the center post and the shroud. Therefore, the 0.5-in. vertical gap left for the center post expansion is adequate to ensure no interference fit between the shroud and the bottom plates of the adjacent baskets.

Radial expansion for the stainless steel plate was evaluated as follows:

$\alpha_{ss} = 9.70 \times 10^{-6} \text{ in/in/}^{\circ}\text{F}$	Stainless steel coefficient of thermal expansion
$\Delta T = 200 \text{ }^{\circ}\text{F}$	Temperature differential
$R_{plate} = 11.31 \text{ in.}$	Bottom plate outside radius
$\Delta L_{ss} = \alpha_{ss} R_{plate} \Delta T = 0.022 \text{ in.}$	Radial expansion of bottom plate.

Radial expansion for the copper shroud was evaluated as follows:

$\alpha_{cu} = 10.0 \times 10^{-6} \text{ in/in/}^{\circ}\text{F}$	Copper coefficient of thermal expansion
$\Delta T = 200 \text{ }^{\circ}\text{F}$	Temperature differential
$R_{shroud} = 11.31 \text{ in.}$	Shroud outside radius
$\Delta L_{cu} = \alpha_{cu} R_{shroud} \Delta T = 0.023 \text{ in.}$	Radial expansion of copper shroud.

The copper shroud is expected to expand radially 0.023 in. while the stainless steel MCO shell and basket bottom plate are expected to expand by approximately 0.022 in. As these values are very comparable, the fabrication gaps will remain open and no undetermined loads will be applied to the MCO shell from unexpected basket expansion loads.

3.2.3.3 Stress Analysis of the Mark IV Fuel Basket. The following section summarizes the design agent's evaluation of the Mark IV fuel basket for lifting and deadweight loading. Section 4.11 of the MCO Performance Specification (Goldmann 2000b) exempts the Mark IV basket from drop accident loading. The structural adequacy evaluation is based on the MCO Performance Specification criteria (Goldmann 2000b) for lifting loading and Section III of the ASME Boiler and Pressure Vessel Code (ASME 1998) for deadweight loading.

The following assumption was used in the evaluation.

- In evaluating the basket center tube for deadweight loading, it is conservatively assumed for that the center tube carries the weight of the baskets above; in evaluating the support rods for deadweight loading, it is conservatively assumed that the support rods carry the full weight of the baskets above.

Other assumptions are as noted within the documentation.

The Mark IV fuel basket geometry is defined in Figure 1-4. The fuel basket's primary structural components are identified in the 60° sector shown in Figure 3-20. The sector shown in Figure 3-20 is repeated every 60°, including the support rod. Holes in the bottom plate are designed to allow drainage from the fuel rods. Each storage basket has a capacity of 54 fuel

assemblies. The components of the Mark IV fuel basket that have been subjected to a structural evaluation are listed in Table 3-27.

Table 3-27. Mark IV Fuel Basket Structural Components.

Component name	Component function
Center tube	(1) Provide support to baskets above when stacked inside the MCO (2) Provide process tube access to the bottom of the MCO (3) Provide lifting grapple interface
Support rod	Provide support to baskets above when stacked inside the MCO
Bottom plate	(1) Provide mounting base for center pipe and support rods (2) Maintain position of spent fuel assemblies after the assemblies are inserted into the basket

The support rods are constructed from 1.3125-in. round bar that is bolted into the bottom plate. The center tube is a hollow bar with a 2.84-in. outer diameter and a 1.75-in. inner diameter. The bar is attached to the bottom plate with a 3.00-8 class 2 thread. The baskets are designed to be stacked within the MCO such that the center tube and six support rods share the weight of the baskets stacked above.

Also present is a 0.05-in.-thick, 11.0-in.-high sheet metal shroud at the basket outer diameter immediately above the bottom plate. The shroud is considered nonstructural.

The material specified for the Mark IV fuel basket components is 304L or 304 stainless steel. For this analysis, the only mechanical properties of interest are the elastic modulus, yield strength, ultimate strength, and allowable stress. The appropriate values were extracted from the ASME Boiler and Pressure Vessel Code, Section II, Part D (ASME 1998). Properties of 304L stainless steel are used.

3.2.3.3.1 Lifting Loads. For lifting loads, the criterion in the MCO Performance Specification (Goldmann 2000b) applies.

3.2.3.3.2 Deadweight Loads. According to Section 4.12.3 of the MCO Performance Specification (Goldmann 2000b), the "baskets will be able to support the fuel at 1.0 g while at 132 °C." The MCO Performance Specification does not specify the acceptance for this loading. For consistency with the Mark IA basket criteria, the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME 1998), was assumed. For membrane and membrane plus bending stresses, the allowable stresses shown in Table 3-28 have been applied.

Table 3-28. Allowable Stresses — Deadweight.

Temperature		S_M	Design/Level A stress limits	
°F	°C		$P_M (S_M)$	$P_M + P_b (1.5 S_M)$
212	100	16.7 ksi	16.7 ksi	25.1 ksi
270	132	16.7 ksi	16.7 ksi	25.1 ksi

Notes: Design limits are from NG-3221 and Level A stress limits are from NG-3222, ASME *Boiler and Pressure Vessel Code*, 1998, American Society of Mechanical Engineers, New York, New York.

Axial compressive stresses must be limited to values established in accordance with one of the following (ASME 1998):

- NG-3133.3 (external pressure)
- NG-3133.6 (axial compression on cylindrical shells)
- NF-3322.1© (column type members).

P_b = bending stress.

P_M = primary membrane stress.

S_M = code allowable stress.

The bottom basket center pipe and support rods are subjected to compression loading, with the potential for column buckling. Since Subsection NG (ASME 1998) does not address column buckling, Subsection NG was supplemented by Subsection NF (ASME 1998). For the center pipe, the more restrictive of NG-3133.6 (shell buckling) or NF-3322.1(c)(2) was used. For the support rods, the Subsection NF criteria were used.

3.2.3.3.3 Stress Analysis Calculations. As previously mentioned, the Mark IV storage baskets were evaluated for two load cases: lifting of the basket and contents with a maximum temperature of 100 °C, and deadweight stacking of the baskets inside the MCO at the design temperature of 132 °C. A full stack consists of five baskets, resulting in the bottom basket supporting the four baskets above it. No other loads were considered. Note that Section 4.11 of the MCO Performance Specification exempts the Mark IV baskets from drop accident loading (Goldmann 2000b).

The Mark IV storage baskets were evaluated using both hand calculations (Mathcad) and finite element calculations (ANSYS). The finite element calculations were limited to stress predictions in the relatively complex bottom plate. The following input parameters were used:

$d_r = 1.31$ in.	Support rod diameter
$l_r = 26.687$ in.	Support rod length extending above bottom plate
$S_y = 19,800$ lb/in ²	304L yield strength at design temperature (132 °C)
$E = 27.2 \times 10^6$ lbf/in ²	304L Young's modulus at design temperature

$D_o = 2.75$ in. Center pipe outer diameter

$D_i = 1.75$ in. Center pipe inner diameter

$l_p = 30.467$ in. Center pipe, overall length

$A_p = \frac{\pi}{4} [D_o^2 - D_i^2]$ Center pipe area (away from grapple interface)

$A_p = 3.534$ in²

$I_p = \frac{\pi}{4} [D_o^4 - D_i^4]$ Center pipe moment of inertia

$I_p = 2.347$ in⁴

$r_p = \frac{1}{4} \sqrt{D_o^2 + D_i^2}$ Center pipe radius of gyration

$r_p = 0.815$

$W_b = 3,218$ lb Mark IV loaded basket weight

$D_t = 3.00 - 8$ Thread attachment, center post to bottom plate 3.0 in. diameter at eight threads per inch.

The following allowable stresses were established for the lifting load condition:

- Criteria
 - Limit stress to one-third material yield or one-fifth ultimate strength
 - Temperature, 100 °C
- Basic tension allowable (304L material)
 - $S_{u100} = 65,600$ lb/in²
 - $S_{u100}/5 = 13,120$ lb/in²
- Base material lifting allowable controlled by yield strength
 - $S_{y100} = 21,000$ lb/in²
 - $S_{y100}/3 = 7,000$ lb/in²

The following allowable stresses were established for the deadweight load condition:

- Assume ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, allowables
- Design temperature, 132 °C (270 °F)
- $S_{m132} = 16,700 \text{ lb/in}^2$
- $S_{y132} = 19,800 \text{ lb/in}^2$

Center Pipe. The following calculations were performed to determine membrane stress on the center pipe under the deadweight stacking load condition.

$A_{P(MIN)}$ = minimum center post area at grapple interface

$$= \frac{\pi}{4} [(2.375 - 0.005)^2 - (2.030 + 0.005)^2] = 1.196 \text{ in}^2$$

Conservatively assuming all four baskets above the bottom basket are carried by the center pipe

$$P_M = \frac{4(W_b)}{A_{P(MIN)}} = \frac{4(3,218)}{1.196} = 10,762 \frac{\text{lb}}{\text{in}^2}$$

$$\text{Ratio} = \frac{P_m}{S_{m132}} = 0.64$$

Therefore, membrane stress is satisfactory.

Buckling was checked in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Paragraph NG-3133.6 (ASME 1998):

$$T = 0.5 \cdot I$$

$$R = \frac{D_o}{2} - T = 0.875 \cdot i$$

$$A = \frac{0.12}{\left(\frac{R}{T}\right)} = 0.071$$

From the ASME Boiler and Pressure Vessel Code, Section II, Figure HA-3 (ASME 1998), $B > 10,000 \text{ lb/in}^2$. The stress through the center portion of post equals

$$\frac{4(W_b)}{A_p} = \frac{4(3,218)}{3.534} = 3,642 \frac{\text{lb}}{\text{in}^2}$$

The calculated ratio is

$$\frac{P_m}{B} = 0.36$$

Therefore, the axial compressive stress limit is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME 1998).

Column buckling was checked in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Paragraph NF-3322.1 (ASME 1998):

$$E = 2.72 \times 10^7 \text{ lb/in}^2$$

$$I_p = 2.347 \text{ in}^4$$

$$K_p = 0.0 \text{ (effective length factor, pinned at top)}$$

$$l_p = 30.467 \text{ in.}$$

$$r_p = 0.815 \text{ in.}$$

$$\frac{K_p l_p}{r_p} = 29.91 < 120$$

$$F_a = S_{y132} \left[0.47 - \frac{\frac{K_l}{r}}{444} \right] = 7,972 \frac{\text{lb}}{\text{in}^2} < B$$

$$\text{Ratio} = \frac{P_m}{F_a} = 0.46 < 1.0$$

Therefore, the limits in the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF are met.

The shear area through the conical lip is taken at a point midway between the outside diameter of the thread and the outside maximum diameter of the conical lip.

$$D_{LIP} = \frac{D_{THREAD} + D_{LIP,OUTER}}{2} = \frac{3.5 + 2.84}{2} = 3.17 \text{ in.}$$

At this point the lip is $0.180 + \left(\frac{3.5 - 3.17}{2} \right) \tan 30^\circ = 0.28 \text{ in.}$ thick, resulting in a shear area of

$$A_{SHEAR} = \pi(3.17)(0.28) = 2.74 \text{ in}^2.$$

Conservatively taking all five baskets (only the base plate and payload weight is carried by the lower basket), the shear stress is

$$P_s = \frac{5(3,218)}{3.93} = 5,869 \frac{\text{lb}}{\text{in}^2}$$

$$\text{Ratio} = \frac{P_s}{0.6(S_M)} = \frac{5,869}{0.6(16,700)} = 0.59$$

The applied load due to deadweight stacking could be carried by the threaded portion of the center pipe-to-base attachment if the thread were loose. The threaded attachment is a 3.00 - 8 class 2A thread with a nominal engagement of one-half the base plate thickness. Allowing for a thread relief at the end of the thread and a chamfer at the top of the base plate, three threads are assumed to carry the load. The shear areas of the external and internal threads are as follows:

$$\frac{A_{S,INTERNAL}}{L_E} = \pi n K_{NMAX} \left[\frac{1}{2n} + 0.57735 (E_{SMIN} - K_{NMAX}) \right]$$

where

L_E = length of engagement (three threads = 0.375 in.)

n = threads per inch (8)

K_{NMAX} = maximum minor diameter of internal thread (2.890 in.)

E_{SMIN} = minimum pitch diameter of external thread (2.9077 in.)

$$\frac{A_{S,EXTERNAL}}{L_E} = \pi (8)(2.890) \left[\frac{1}{2(8)} + 0.57735(2.9077 - 2.890) \right] = 5.282 \frac{\text{in}^2}{\text{in}}$$

$$\frac{A_{S,INTERNAL}}{L_E} = \pi n D_{SMIN} \left[\frac{1}{2n} + 0.57735 (D_{SMIN} - E_{NMAX}) \right]$$

where

D_{SMIN} = minimum major diameter of external thread (2.9824 in.)

E_{NMAX} = maximum pitch diameter of internal thread (2.9299 in.)

$$\frac{A_{S,INTERNAL}}{L_E} = \pi (8)(2.9824) \left[\frac{1}{2(8)} + 0.57735 (2.9824 - 2.9299) \right] = 6.957 \frac{\text{in}^2}{\text{in}}$$

Therefore, the external thread governs the strength of the connection.

$$A_{S,EXTERNAL} = (5.282)(0.375) = 1.981 \text{ in}^2$$

Conservatively taking all five baskets (only the base plate and payload weight is carried by the lower basket), the total deadweight stacking load is

$$5(W_b) = 5(3,218) = 16,090 \text{ lb}$$

The thread shear stress is

$$F_{ST} = \frac{P_M}{A_{S,EXTERNAL}} = \frac{16,090}{1.981} = 8,122 \frac{\text{lb}}{\text{in}^2}$$

$$\text{Ratio} = \frac{P_M}{0.6 S_{M132}} = \frac{8,122}{0.6(16,700)} = 0.81$$

The lifting load condition was evaluated as follows:

- Section through grapple interface

$$A_{p(min)} = 1.196 \text{ in}^2$$

$$P_{m(lifting)} = \frac{W_t}{A_{p(min)}} = \frac{3,218}{1.196} = 2,691 \frac{\text{lb}}{\text{in}^2}$$

$$\text{Ratio} = \frac{P_m}{\frac{1}{3} S_y} = \frac{2,691}{7,000} = 0.38$$

- Section through conical lip

$$P_{sl} = \frac{W_t}{A_{Lip}} = \frac{3,218}{2.74} = 1,174 \frac{\text{lb}}{\text{in}^2}$$

$$\text{Ratio} = \frac{P_{SL}}{0.6 \left(\frac{1}{3} S_y \right)} = \frac{1,174}{0.6(7,000)} = 0.28$$

- Shear stress in threaded attachment

$$P_{ST} = \frac{W_t}{A_{thread}} = \frac{3,218}{1.981} = 1,624 \frac{\text{lb}}{\text{in}^2}$$

$$\text{Ratio} = \frac{P_{ST}}{0.6 \left(\frac{1}{3} S_y \right)} = \frac{1,624}{0.6(7,000)} = 0.39$$

Support Rod. The support rods are loaded in compression by the deadweight stacking load:

$$d_r = 1.31 \text{ in.}$$

$$I_r = \frac{\pi(d_r)^4}{64} = 0.145 \text{ in}^4$$

$$K_r = 2.1 \quad \text{Effective length factor, fixed-free}$$

$$l_r = 26.687$$

$$r_r = \frac{d_r}{4} = 0.33 \text{ in.}$$

$$A_r = 1.348 \text{ in}^2.$$

Using ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Paragraph NF-3322.1(c)(2) (ASME 1998)

$$\frac{K_r l_r}{r_r} = 169.8 > 120$$

$$F_s = S_{y132} \left[0.40 - \frac{K_r l_r / r_r}{600} \right] = 2,315 \frac{\text{lb}}{\text{in}^2}$$

Conservatively assuming the bottom basket support rods carry the full weight of the four baskets above

$$F_r = \frac{4(W_b)}{6(A_r)} = 1,591 \frac{\text{lb}}{\text{in}^2}$$

$$\text{Ratio} = \frac{F_r}{F_s} = 0.69$$

The ratio is less than 1.0, therefore the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, are met for the support rods.

The support rods are attached to the base plate with .625-11 flat head screws. Torque is applied in accordance with the limits of the Boiler and Pressure Vessel Code, Subsection NG-3232.2 (ASME 1998), the preload stress being limited to $(1.2)(0.9 S_Y)$. For SA-193 material at 270° F

$$S_Y = 3(S_M) = 3(11,600) = 34,800 \text{ lb/in}^2$$

The stress limit is therefore

$$(1.2)(0.9)(34,800) = 37,584 \text{ lb/in}^2$$

The maximum allowed preload force is therefore

$$F = \sigma A_s = 8,494 \text{ lb}$$

The stripping of the thread in the support rod is checked for this loading as follows:

$$\frac{A_{S,INTERNAL}}{L_E} = \pi n D_{SMIN} \left[\frac{1}{2n} + 0.57735 (D_{SMIN} - E_{NMAX}) \right]$$

where

D_{SMIN} = minimum major diameter of external thread = 0.6113 in.

E_{NMAX} = maximum pitch diameter of internal thread = 0.5732 in.

L_E = length of engagement

= 2.50 (length of bolt) - 1.20 (maximum thickness of base plate)

= 1.30 in.

$$\frac{A_{S,INTERNAL}}{L_E} = \pi (11) (0.6113) \left[\frac{1}{2(11)} + 0.57735 (0.6113 - 0.5732) \right] = 1.4125 \frac{\text{in}^2}{\text{in}}$$

$$A_{S,INTERNAL} = (1.425) (1.30) = 1.85 \text{ in}^2.$$

The allowable stress is

$$0.6 S_M = 0.6(16,700) = 10,020 \text{ lb/in}^2.$$

The stripping load is

$$(1.85)(10,020) = 18,500 \text{ lb} > 8,494 \text{ lb preload}$$

Therefore stripping of the threads is not a concern.

With a thread friction coefficient of $k = 0.2$, the calculated torque for $d = 0.625$ in. is $T = Fkd/12 = 88 \text{ lb-ft}$. With an uncertainty of 30%, the torque should be limited to 68 lb-ft. The recommended torque is $60 \pm 8 \text{ lb-ft}$.

The use of a clamped condition for the attachment of the support rod to the base plate is substantiated by applying the load at which the rod buckles to the extreme edge of the rod and ensuring the preload in the bolt is higher than the load induced by the resulting moment around the attachment interface of the rod to base plate (see Figure 3-21). Using the information in Table 3-29

$$Y = \frac{\Sigma AY}{\Sigma A} = \frac{0.1816}{0.4577} = 0.3966 \text{ in}$$

$$I = \Sigma AY^2 + \Sigma A(Y)^2 = 0.0963 - (0.4577)(0.3966)^2 = 0.0243 \text{ in}^4$$

Table 3-29. Mark IV Rod Geometric Properties.

Section	Area	Y	AY	AY ²
1	0.3447	0.2331	0.0803	0.0187
2	-0.1130	0.4135	-0.0467	-0.0193
3	0.226	0.6550	0.1480	0.0969
$\Sigma \rightarrow$	0.4577	--	0.1816	0.0963

The moment at the support rod to base plate interface with the applied load equal to the buckling load is

$$\begin{aligned}
 M &= (\text{buckling stress}) \times (\text{rod area}) \times (\text{edge of rod to neutral axis}) \\
 &= 2,315 \times 1.348 \times 0.3966 \\
 &= 1,238 \text{ in-lb} .
 \end{aligned}$$

The applied load to the bolt is

$$P_{\text{BOLT}} = (Mc / I)(A_{\text{BOLT}})$$

where

c = distance from bolt centerline to neutral axis.

$$P_{\text{BOLT}} = [1,238 (0.655 - 0.3966) / 0.0243] (0.226) = 2,975 \text{ lb} .$$

The minimum preload in the bolt is determined using the minimum applied torque and a 30% uncertainty factor:

$$\begin{aligned}
 \text{Minimum preload} &= [(60 - 8)(12) / (0.2)(0.625)] (1 - 30\%) \\
 &= 3,840 \text{ lb} > 2,975 \text{ applied load.}
 \end{aligned}$$

The connection satisfies the analytic assumption of a fixed end.

Bottom Plate. The bottom plate was evaluated for both the deadweight stacking condition inside the MCO and the lifting condition. These load conditions are evaluated in the subsections that follow.

The critical bottom plate for the deadweight stacking load condition is the second basket from the bottom with the bottom basket support rods rotated 30° relative to the support rods for the basket immediately above. It was conservatively assumed that the weight of the top three baskets is carried entirely by the support rods of the fourth basket. This configuration develops

the maximum bending stress in the bottom plate. For this load condition, the fourth basket support rods would each carry one-sixth of the weight of the top three baskets (single basket weight is 3,212 lb [Goldmann 2000b]):

$$F_r = (3 \times 3,212) / 6 = 806 \text{ lb}.$$

The force is applied as an equivalent pressure over the area of the support rod in contact with the plate. The contact area and pressure are as follows

$$\text{Area} = 1/4 [(1.313)^2 - (0.502)^2] \pi = 1.16 \text{ in}^2$$

$$\text{Pressure} = 806/\text{area} = 695 \text{ lb/in}^2.$$

In addition to the loading on the support rods, the bottom plate also carries the weight of the fuel rods in the basket. The fuel rod weight was applied as an equivalent pressure (single rod weight is 55.4 lb [Goldmann 2000b]):

$$\begin{aligned} \text{Area} &= 1/4 [(22.625)^2 - (3.00)^2] \pi - 1/4 (0.51)^2 (108\pi) - 1/4 (1.313)^2 (6\pi) \\ &= 364.8 \text{ in}^2 \end{aligned}$$

$$\text{Pressure} = 54(55.4)/\text{area} = 8.20 \text{ lb/in}^2.$$

The equivalent pressure approach was judged to be conservative because it moves the center of loading radially outward, relative to the actual fuel support locations, resulting in higher bending moments at the maximum stress locations (outer ligaments). Because of the 30° symmetry, only one-twelfth of the basket was modeled.

For the deadweight stacking load condition, the loading comprised a 695 lb/in² downward vertical pressure on the area of the plate in contact with the support rod, an 8.20 lb/in² downward pressure on the remainder of the bottom plate, and a 1.0 g deadweight acceleration. The model was constrained vertically at the bottom outer corners of the support rod bolt hole (adjacent basket, below, support rod locations, 30° rotation).

The deadweight stacking load case results show the maximum stress intensity is 19,180 lb/in². The higher stress intensity values occur near the support rod bolt hole on the top of the plate. The allowable primary membrane plus bending stress intensity is 25,100 lb/in², resulting in the following stress ratio for the deadweight stacking load case:

$$19,180 / 25,100 = 0.76.$$

The same ANSYS model was used for the lifting load case, but with boundary condition changes. The only constraint was a vertical constraint inside radius of the bottom plate (where the center pipe is threaded to the plate). The loading included 1.0 g gravity loading and the 8.20 lb/in² downward pressure representing the gravity loading from the fuel. A maximum stress intensity of about 3,529 lb/in² occurred on the top of the plate at the inside radius of the plate. This maximum ligament stress is essentially all primary bending stress and is less than the

allowable lifting stress of 7,013 ksi. The stress ratio for the lifting load case is $3,529/7,013 = 0.50$.

Load Distribution and Basket Interface Considerations. When the baskets are stacked inside the MCO, the center pipe and support rod load distribution is very sensitive to the interface geometry. Reasonable fabrication tolerances will result in an imperfect fit, which will likely result in a three-point contact at the basket interface (three rods or two rods and the center pipe). There is also the possibility that the center pipe will carry the entire load.

Even for the case of a perfect fit, the stack loading will not be evenly shared between the center pipe and rods. Using the ANSYS model, the perfect fit load distribution was calculated by applying the fuel deadweight pressure to the bottom plate and obtaining the reactions at the support rod and center pipe. From the ANSYS output, the support rod and center pipe reactions were 397 lb and 137 lb, respectively (534 lb total). The center pipe load ratio is $137/534 = 0.26$. However, for the imperfect fit, the center pipe load ratio could range from zero to one.

In order to establish reasonable tolerances on the basket interface dimensions, capacity force/deflection response predictions were made for both a support rod and the center pipe. By knowing the force/deflection response, the effect of component length differences on the component load sharing can be evaluated. The capacity force/deflection response was obtained using the ANSYS plastic beam element (BEAM23), with large deflections/strain enabled. Buckling was initiated by assuming a 0.25-in. offset of the vertical load.

Assuming that three support rods are supporting four baskets above, the force per rod is $4(3,177)/3 = 4,236$ lb. A deflection of about 0.070 in. is achieved in a support rod before the load capacity drops below 4,236 lb. A less conservative force/deflection rod response was obtained by using a gap/friction element on top of the rod to account for lateral constraint due to friction. Using a conservative friction coefficient of 0.1, a much higher capacity and deformation was obtained when rod frictional constraint was considered.

Assuming that the bottom center pipe carries the full load from the four baskets above, the center pipe loading is $4(3,177) = 12,710$ lb. The center pipe capacity is well in excess of the loading. Also note that with a 0.125-in. center pipe deflection, the plastic buckling mode has not been reached.

The above force/deflection results indicate that tight basket interface tolerances are not necessary. Conservatively using half of the 4,236 lb deflection of 0.070 in., fabrication height differences of 0.035 in., or smaller, are acceptable. It appears that the interface fabrication tolerance issue may be controlled by functional, rather than structural, considerations. For example, a 0.035-in. height difference between the center pipe and support rods results in a horizontal tipping distance of about 0.10 in.

Component Stress Results Summary. From the calculations above, a summary of the component stress analysis results was compiled and is presented in Table 3-30. Note that the predicted stresses are below allowables for all components.

Table 3-30. Summary of Mark IV Basket Structural Evaluation.

Item	Critical load condition	Stress category	Maximum stress (lb/in ²)	Allowable	Ratio
Center pipe	Dead weight stacking	Buckling	3,642	7,972	0.46
Conical lip	Dead weight stacking	Pure shear	5,869	10,020	0.59
Center pipe plate threads	Dead weight stacking	Pure shear	8,122	10,020	0.81
Support rod	Dead weight stacking	Buckling	1,591	2,315	0.69
Bottom plate	Stacking	Pm+Pb	19,180	25,100	0.76
Bottom plate	Lifting	Pm+Pb	3,529	7,013	0.50

Pb = bending stress.

Pm = primary membrane stress.

3.2.4 Multi-Canister Overpack Shield Plug Interface Components, Canister Collar, Locking Ring, and Cover Cap

This section summarizes the design agent's evaluation of the MCO shield plug interface components for preload torquing of the cover plate bolts and process valve bodies, design pressure, and drop loading, and the evaluation of the canister cover and cover cap for lifting and pressure loading.

3.2.4.1 Shield Plug Interface Components. This section summarizes the design agent's evaluation of the MCO process port cover plates, cover plate bolts, process valves (including the rupture disk), and the process filter attachment welds. The structural adequacy evaluation is based upon Subsection NB of Section III of the ASME Boiler and Pressure Vessel Code (ASME 1998). Component loading includes preload from torquing of bolts and process valve bodies, design pressure, and drop loading.

The following assumptions were used in the evaluation.

- Preloads from torquing the cover plate bolts and process valves are accurate to within +30% and -35%. This range is assumed to include the standard preload/torque uncertainties, including operator and tool inaccuracies.
- The maximum temperature during torque application (insertion and removal) is the design temperature of 132 °C. It is assumed that the nut factors are not affected by temperature (Bickford 1990). The adequacy of the uncertainty range and the mean nut factor for the cover plate bolts and process valves must be verified by test.

- The threaded process valve bodies can be appropriately evaluated as bolts, relative to the ASME Boiler and Pressure Vessel Code design stress limits.
- The torquing tool used for inserting and preloading the rupture disk process plug will extend beyond the holes in the hex head. That is, the hex head minimum cross section, at the vent holes, will not experience the full torque during torquing operations.

Other assumptions are noted within the calculation documentation.

Some of the dimensions of the rupture disk valve (e.g., rupture disk outside diameter) are based upon dimensions provided by a specific rupture disk manufacturer. Since the dimensions are vendor-dependent, some of the calculations may need to be modified, depending on the final rupture disk manufacturer selected.

The geometry of the shield plug interface components is defined on the assembly drawing (Figure 1-1), the shield plug drawing (Figure 1-8), the process plug valve drawing (Figure 1-10), the cover plate and bolts drawing (Figure 1-11), and the process filter drawing (Figure 1-12). The structural components are identified in Figure 3-22. There are three process ports and one plugged port in the shield plug. The structural components of the port closures are identical, except that three cover plates on each MCO have four bolts and one has five. Only the weaker four-bolt configurations have been evaluated. The purpose of the cover plates is to provide secondary containment for the process valve seals, and to protect the valves during handling operations.

Each process valve body has a hex head used for torquing the threaded valve bodies into the four shield plug port holes. Sealing for both the process valves and the cover plates is achieved using a C-seal requiring a minimum seating load of 300 lb/in. The plugged port has a threaded portion (1-1/16 12 thread) that is smaller than the 1.875-in. process valve threads.

The process filter is welded to the bottom of the shield plug. Only the 0.125-in. welds attaching the filter to the shield plug have been evaluated by the design agent (Goldmann 2000a). A bounding filter assembly weight of 50 lb is specified by the MCO Performance Specification (Goldmann 2000b).

The materials included in the shield plug interface components are listed in Table 3-31. The structural properties of interest for F304L stainless steel are provided in Table 3-32 and properties for type 304L stainless steel are provided in Table 3-33. The cover plate bolt and process valve material properties at the MCO design temperature of 132 °C are listed in Table 3-34.

The process valves are constructed from SA-193, grade B8S or B8SA. Since the valve bodies function much like a bolt (external threads, hex head, provide preload to seal), they have been evaluated using ASME rules for bolts. The process filters will be welded to the 304L shield plugs. Therefore, the 304L base material allowable limits, Table 3-33, have been applied to the attachment welds.

Table 3-31. Material Listing for Shield Plug Interface Components.

Component	Material	ASME Spec No.
Shield plug, process port cover plate	304L	SA-182
Cover plate bolts	18Cr-8Ni-4Si-N	SA-193-B8S or B8SA
Process valves	18Cr-8Ni-4Si-N	SA-193-B8S or B8SA
Process port cover plate	304L	SA-240 or SA-479

Table 3-32. Material Properties for Type F304L Stainless Steel (SA-182, $t > 5$ in.).

Temperature		E	S_M	S_Y	S_U
°F	°C	Table TM-1, Group G	Table 2A, page 322	Table Y-1, page 524	Table U, page 441
70	--	28.3 E+06	16.6 ksi	25.0 ksi	65.0 ksi
100	--	--	16.6 ksi	25.0 ksi	65.0 ksi
200	--	27.6 E+06	16.6 ksi	21.3 ksi	61.5 ksi
212	100	27.5 E+06	16.6 ksi	21.0 ksi	60.9 ksi
270	132	27.2 E+06	16.6 ksi	19.8 ksi	58.0 ksi
300	--	27.0 E+06	16.6 ksi	19.1 ksi	56.5 ksi

Note: Underlined values have been determined by linear interpolation, all other values have been taken from Section II, Part D of the ASME Boiler and Pressure Vessel Code.

E = material modulus of elasticity.

S_M = material allowable stress.

S_U = material ultimate strength.

S_Y = material yield strength.

t = material thickness.

Table 3-33. Material Properties for Type 304L Stainless Steel
(SA-240 or SA-479).

Temperature		E	S _M	S _Y	S _U
°F	°C	Table TM-1, Group G	Table 2A, page 322	Table Y-1, page 524	Table U, page 441
70	--	28.3 E+06	16.7 ksi	25.0 ksi	70.0 ksi
100	--	--	16.7 ksi	25.0 ksi	70.0 ksi
200	--	27.6 E+06	16.7 ksi	21.3 ksi	66.2 ksi
212	100	27.5 E+06	16.7 ksi	21.0 ksi	65.6 ksi
270	132	27.2 E+06	16.7 ksi	19.8 ksi	62.5 ksi
300	--	27.0 E+06	16.7 ksi	19.1 ksi	60.9 ksi

Note: Underlined values have been determined by linear interpolation, all other values have been taken from Section II, Part D of the ASME Boiler and Pressure Vessel Code.

E = material modulus of elasticity.
S_M = material allowable stress.
S_U = material ultimate strength.
S_Y = material yield strength.

Table 3-34. Material Properties for Cover Plate Bolts and Process Valves at
Multi-Canister Overpack Design Temperature.

Material	Elastic modulus, lb/in ² (270 °F)	S _m , lb/in ² (270 °F)
SA 193 Grade B8S or B8SA (Bolting)	27.2 E+06	11,600

Notes: All values are based on Section II, Part D, of the ASME Boiler and Pressure Vessel Code. For F304L, 304L, SA193 Grade B8S or B8SA materials, the mean coefficient of thermal expansion from 70 °F to 270 °F is 8.94 x 10⁻⁶ in/in/°F (ASME Boiler and Pressure Vessel Code, Table TE-1).

The requirements in the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB (ASME 1998), are used as the acceptance criteria for the shield plug interface components. All of the interface components are part of the MCO pressure boundary with the exception of the process filter attachment welds. For the process filter attachment welds, it is assumed that the inspection will be limited to a visual examination and appropriate weld quality factors from the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME 1998), are applied. The full design temperature allowable limits are conservatively applied to the attachment weld evaluation.

A rupture disk is included in one of the process valve designs, which results in a two-piece design assembled with a seal weld at the edge of the disk and a structural weld connection. The structural weld is subjected to both torque loading and the design pressure. The weld is performed using an electron beam process weld. The Subsection NG weld quality factors (ASME 1998) are applied to the rupture disk valve structural weld, assuming a surface-only dye penetrant examination. The resulting design margin is large.

It is expected that the initial attachment of the process valves and cover plates to the shield plug will be conducted at room temperature. For later processing operations, torquing of both the cover plate bolts and process valves may occur at higher temperatures. To cover the possibility of torquing operations at higher temperatures, it has conservatively been assumed that torquing could occur at the full design temperature of 132 °C. Another conservative assumption is that the maximum torque to the process valves could occur simultaneously with the full design pressure (e.g., when the breakaway torque is applied during valve opening).

The critical loading for the process valves and cover plate bolts is the torquing preload combined with the design pressure and temperature. Because of the relative sizes of these components, the vertical drop inertia loading is less than the pressure loading. For the more severe horizontal drop, the parts bear against the sides of shield plug holes (i.e., the cover plate sits in a recess). Appendix F of the ASME Boiler and Pressure Vessel Code (ASME 1998) does not limit bearing stresses. Thus, only the filter-attachment welds are evaluated for the drop loading.

Table 3-35 provides a summary of the evaluation results. See the MCO Design Report, Appendix 10 (Goldmann 2000a), for further detail on the structural evaluation of the shield plug interface components.

3.2.4.2 Canister Collar and Cover Cap. This section summarizes the design agent's evaluation of the cover cap and canister collar under lifting and pressure loads (Goldmann 2000a, Appendix 6). The drop load evaluation is summarized in Section 3.2.2. This section also documents the evaluation of the weld at the cover cap-canister collar interface. The evaluations have been performed based on the criteria in the ASME Boiler and Pressure Vessel Code (ASME 1998). A combination of hand calculations and ANSYS analysis has been used for preparing the evaluation.

The canister collar is welded to the shell to provide a sealing surface for the shield plug seal. The canister collar has a double lead buttress thread for positioning the locking ring. After cold vacuum drying of the MCO is completed and before the MCO is placed into interim storage, a cap is placed over the shield plug and welded to the canister collar. This cover cap is designed to accommodate a lifting grapple with six gripping shoes. The cap includes a plug and welded cover plate at its center for leak testing and helium backfilling.

Table 3-35. Summary of Structural Evaluation of Shield Plug Interface Components.

Item	Critical load condition	Stress category	Maximum stress (lb/in ²)	Allowable stress (lb/in ²)	Ratio
Process cover plate bolts	Pressure and preload	In-service	20,720	37,584	0.55
	Pressure and preload	Appendix E of ASME Code	Required area = 0.674 in ²	Actual area = 0.756 in ²	0.89
	Thread shear	Bolt allow	6,278	11,880	0.53
Cover plate	Pressure	Bending	2,665	24,900	0.106
Process valve plug	Pressure and preload	Appendix E of ASME Code	Required area = 0.284 in ²	Actual area = 0.981 in ²	0.29
	Pressure and preload	In-service	6,230	23,200	0.268
	Torque and in-service	In-service	7,404	23,200	0.32
	Hex head torque	In-service	18,890	23,200	0.81
Rupture disk process valve	Torque and in-service	In-service shear	4,394	5,570	0.79
Process filter attachment welds	101 g horizontal	Level D shear	3,133	3,507	0.893

The cover cap is fabricated from SA-182 F304L stainless steel, and the canister collar is fabricated from SA-182 (dual certified) type F304L/F304 stainless steel. Material properties are taken from Section II, Part D, of the ASME Boiler and Pressure Vessel Code (ASME 1998) and are listed in Table 3-36.

According to Section 4.11 of the MCO Performance Specification (Goldmann 2000b), the cover cap design "shall be capable/configured for lifting the MCO with the same equipment described in Section 4.13." Section 4.13 of the MCO Performance Specification (Goldmann 2000b) describes such equipment as a "lifting ring with a 12 ton lifting capacity." Therefore the cover cap shall have a lifting capacity of 12 ton. Furthermore "the lifting ring design and cover cap lifting rim area must exhibit a safety factor of three on material yield and five on material ultimate strength" as instructed in the MCO Performance Specification (Goldmann 2000b).

Table 3-36. Material Properties for Cover Cap and Canister Collar.

SA-182 F304L stainless steel forging				
	70 °F	200 °F	300 °F	270 °F
E - psi	28.3 E+06	27.6 E+06	27.0 E+06	27.18 E+06
S _M - psi	16,700	16,700	16,700	16,700
S _Y - psi	25,000	21,300	19,100	19,760
S _U - psi	70,000	66,200	61,500	62,910
Mean coefficient of thermal expansion from 70° to temperature indicated - in/in/°F x 10 ⁻⁶				
	70 °F	250 °F	300 °F	270 °F
α-in/in/°F	8.46 E+06	8.90 E+06	9.00 E+06	8.94 E+06
SA-182 F304 stainless steel forging				
	70 °F	200 °F	300 °F	270 °F
E - psi	28.3 E+06	27.6 E+06	27.0 E+06	27.18 E+06
S _M - psi	20,000	20,000	20,000	20,000
S _Y - psi	30,000	25,000	22,500	23,250
S _U - psi	75,000	71,000	66,000	67,500
Mean coefficient of thermal expansion from 70° to temperature indicated - in/in/°F x 10 ⁻⁶				
	100 °F	200 °F	300 °F	270 °F
α-in/in/°F	8.55 E+06	8.79 E+06	9.00 E+06	8.97 E+06

Note: Values have been taken from Section II, Part D, of the ASME Boiler and Pressure Vessel Code.

E = material modulus of elasticity.

S_M = material allowable stress.

S_U = material ultimate strength.

S_Y = material yield strength.

The MCO Performance Specification requires that "the cap shall be capable of withstanding the pressure rating of 450 lb/in² gauge at 132 °C" (Goldmann 2000b, Section 4.11). The MCO Performance Specification (Goldmann 2000b) does not provide criteria for the cover cap and canister collar under these loads, thus the normal (service level A) condition criteria of Subsection NB of Section III of the ASME Boiler and Pressure Vessel Code (ASME 1998) have been used.

According to Section 4.17 of the MCO Performance Specification (Goldmann 2000b), "All MCO pressure boundary welds and welds bearing the fully loaded MCO must be designed for and pass 100% volumetric examination (x-rays or ultrasonic) per ASME requirements" and "All MCO

fabricator pressure boundary welds shall be made in accordance with ASME Section III, Division I, NB-3350." Therefore the stress limits for a full-penetration groove weld shall not exceed the stress values for the base metal being joined.

The cover cap must support the total weight of the MCO and contents for lifting, which equates to a conservative total lifting capacity of 12 ton. A lifting grapple with six gripping shoes will be used to lift the MCO and its contents by the cover cap. Figure 3-23 displays the gripping shoe configuration for the lifting grapple-cover cap interface. The stress through the MCO shell due to the lifting load is 615 lb/in². The thinnest point in the shell is located at the thread relief in the canister collar. Since it will also see the lifting load through its section, the thread relief was analyzed and found to have a stress of 786 lb/in².

The cover cap must be able to withstand an internal pressure of 450 lb/in² gauge. The MCO shell has a thickness of 0.5 in. and its inside diameter is 23.00 in. The stress through the shell due to the pressure load was found to be 10,575 lb/in². The stress for the thinnest section through the canister collar buttress thread (0.373 in. thick) is 15,024 lb/in².

When both lifting and pressure loads are applied together, the stress through the thinned portion of the cover cap shell due to these loads then becomes

$$\text{Axial stress} = 22,687 \text{ lb/in}^2$$

$$\text{Axial stress ratio} = 0.97.$$

$$\text{Hoop stress} = 4,673 \text{ lb/in}^2$$

$$\text{Hoop stress ratio} = 0.2.$$

The cover cap and canister collar also were evaluated using finite element methods. A 180° section of the cover cap and canister collar was modeled using ANSYS finite element analysis. Symmetric boundary conditions were applied at the edges of the models. The bottom edge was fixed in the vertical direction to approximate the lifting configuration. Two loads were applied to the models: one to develop the stresses associated with pressure, and the other to develop the stresses associated with lifting. The 450 lb/in² gauge design pressure was applied uniformly to the inside of the model. In order to apply the lifting load properly, the model was built using a series of slices of different thicknesses and refined mesh in the appropriate areas. The intent of this was to apply the lifting load on the few elements that represent the area of a gripping shoe. According to Section 4.13 of the MCO Performance Specification (Goldmann 2000b), the MCO will be lifted by "...six equally spaced 1.97 inch tangential length by 0.66 inch radial contact length grippers." The area of the gripper was approximated to be 1.3 in² (1.97 in. x 0.66 in.). The finite elements used for this analysis report stresses at the location of the gripping shoe, at the canister collar-lifting cap weld, and at the rupture disk location.

Table 3-37 compares the calculated results of hand calculations to the ANSYS analysis results. Stress results for several locations on the assembly are summarized and ratioed to the allowable limits in Table 3-38.

Table 3-37. Hand versus ANSYS Results.

Stress	Stress category	Hand calculation results (lb/in ²)	ANSYS results (lb/in ²)
S_L	P_M	615	762
S_{HOOP}	P_M	10,575	10,060
S_{L+P}	P_M	5,903	5,460

Table 3-38. ANSYS Results for Pressure Plus Lifting.

Location	Criteria	Stress intensities		Ratio
		Maximum (ksi)	Allowed $P_M + P_B$ (ksi)	
Lifting ear	Goldmann 1998	9.95	19.76	0.5
Lifting cap mid-radius	Goldmann 1998	13.45	19.76	0.68
Weld	ASME Section NG	10.27	25.1	0.41
Bottom of collar*	ASME Section NG	6.9	34.9	0.20

Note: Lifting load of three times the design load is applied in the analysis. All stress results include a 450 lb/in² internal pressure.

*Although it is not conservative to include pressure, the effect is small.

The locking ring thread design involves buttress threads with a specified pitch of 0.25 in. and an engagement length of 2.770 in. The buttress thread design details are shown in Figure 3-24 and are in conformance with ANSI B1.9-1973, *Buttress Inch Screw Threads — 7°/45° Form with 0.6 Pitch Basic Height of Thread Engagement* (ANSI 1973). Because of the importance of the buttress closure threads in providing and maintaining the MCO seal loading, a detailed analysis of the threads and other closure hardware has been performed by the design agent (Goldmann 2000a). The detailed analysis approach, results, and conclusions follow.

As an initial check on thread adequacy for the bolt loading, thread stripping (thread shear) calculations were performed on the locking ring threads. ANSI B1.9-1973 (ANSI 1973) does not

specify a formula for external thread stripping area, so a conservative approximation from *Machinery Handbook* (Green and McCauley 1996) was used. A selected stripping/shear area, per inch of engagement, resulted in a stripping area of 105.16 in² (Green and McCauley 1996, page 1718).

The locking ring is made of type 304N stainless steel. At a maximum design temperature of 132 °C, the maximum bolt in-service loading is 200,000 lb. The thread shear stress results are as follow (allowable thread stress is $0.6S_m$ [ASME 1998, NB-3227.2]):

- Shear stress = 1,902 lb/in²
- Allowable stress = $0.6S_{m132} = 13,644$
- Stress ratio = 0.139

Thus, the thread stripping area is adequate for the maximum set screw loading.

The following paragraphs address the stress analysis of the worst-case lifting condition for the MCO. It is assumed that the MCO is at its maximum temperature, the set screw and seal loads are still applied, the MCO is full of water so it is at its maximum weight, and the maximum internal pressure of 450 lb/in² gauge has leaked past the seal and is applied to the cover cap. It is required that the resulting membrane stress be less than S_m for the materials involved.

The lifting load path proceeds from the cover cap, down through the collar to the shell. The most highly stressed portion of the load path is at the thread relief area in the collar where the wall is a minimum of 0.373 in. thick. The applied loads are as follow:

- MCO weight = 24,000 lb
- Temperature = 270 °F
- Internal pressure = 450 lb/in²
- Jack bolt load = 200,000 lb.

The pressure load of 207,839 lb is applied over the internal diameter of the cap just above the collar threads.

Thus the total axial load is

$$24,000 \text{ lb} + 207,839 \text{ lb} + 200,000 \text{ lb} = 431,839 \text{ lb.}$$

The resulting membrane stress from this load is 14,784 lb/in².

For a collar material of SA-182 type F304L/F304 stainless steel at 270 °F, the allowable stress is 31,940 lb/in². Thus the thread relief area of the collar meets the requirement that the membrane stress, 14,784 lb/in², be less than the allowable stress.

3.2.5 Materials Selection and Corrosion Resistance

3.2.5.1 Materials Selection. Austenitic stainless steel will be used for the majority of the MCO components. Aluminum and copper alloys will be used for some MCO components.

3.2.5.1.1 Austenitic Stainless Steel. The primary reasons for selecting austenitic steel rather than ferritic steel are austenitic steel's corrosion resistance, lower costs for material testing and fabrication, and the availability of material with suitable properties. Austenitic stainless steel is compatible with the 304L stainless steel MCO shell and fuel baskets. Other significant advantages of using austenitic stainless steel are the reduced maintenance required (especially during storage prior to handling) and fewer corrosion protection requirements for threaded and seal surfaces compared with ferritic materials.

The disadvantage of austenitic stainless steel is reduced strength. The lower allowable strength of the material results in a lower margin beyond that inherent in the allowable stresses (ASME 1998). However, austenitic stainless steels are inherently tough and plastically deform, absorbing a great deal of energy before catastrophic failure. Also, all of the construction materials should be of the same basic type to preclude differential thermal expansion and challenges to the sealing system. Galling of bolted connections, which could result in excessively high torque values and/or insufficient preload, is not expected.

Designs of bolted connections in which austenitic materials are used typically include a minimum differential hardness. This may be achieved by varying the cold work that the parts are subjected to during fabrication or by specifying different materials that have inherently differing hardnesses. The locking ring is fabricated out of 304N or 304H stainless steel, reducing the galling potential for both the set screws and the main buttress threads. The process valves, set screws, and cover plate bolts are fabricated from Nitronic 60 (UNS S21800) stainless steel to provide harder surfaces that further minimize the potential for galling. Galling also can be prevented by using smoother surface textures, coarser threads, slower wrenching speeds, and, most importantly, good thread lubrication (Bickford 1990). There are few restrictions on the use of lubricants on any of the threaded fasteners used on the MCO because none of the fasteners are in the pool at any time. Also, all of the fasteners except the process port valves are outside the pressure boundary. The lubricants must still be functional and not offgas after being heated to a minimum of 132 °C (270 °F). With this in mind, high quality, nuclear grade lubricants, such as Nickel Never Seize⁷ or Fel-Pro⁸ Nickel 5000 Never Seize, may be used on threaded surfaces. If some components are placed in the pool, lubricants such as NeoLube⁹ (graphite based) may be used. If continued lubricity is needed after contact with the pool water, it is recommended that the components be suitably relubricated.

⁷Never-Seize is a trademark of USM Corporation.

⁸Fel-Pro is a trademark of Fel-Pro Incorporated.

⁹NeoLube is a trademark of Huron Industries.

3.2.5.1.2 Other Materials. Aluminum and copper are the only other major materials used other than the various grades of stainless steel in the MCO and fuel baskets. Aluminum-based alloys are used for the fuel rack insert and spacer on the fuel baskets to provide a positioning grid during the fuel loading. The major requirement is that the material not interfere with the processing of the fuel and that it not lose sufficient strength such that it will block any of the gas flow passages in the baskets. A detailed evaluation of the initial cast alloy chosen, ASTM B26 356.0-T6, can be found in HNF-SD-SNF-ER-018, *Evaluation of Cast Carbon Steel and Aluminum for Rack Insert in MCO Mark IA Fuel Basket* (Graves 1997a). The current design calls for fabrication of the spacer from 5005H-34 plate and the rack insert from either 6061-T6 or A03560-T6 plate. The greater thermal expansion rate of aluminum (compared to stainless steel) is considered in the component tolerances. Upon expansion, the aluminum fuel rack insert would deform before any significant deformation of the basket shroud occurred. However, after the fuel is loaded, neither the basket shroud nor the fuel rack insert has any significant role in the safety or storage of the fuel.

A wrought copper alloy (C12200) is employed for the divider (shroud) subassembly in the Mark IA and Mark IV scrap baskets because it increases the amount of heat conduction from the fuel fines area, thus increasing safety margin during cold vacuum drying. Copper's thermal expansion coefficient is only about 5% to 10% larger than that of stainless steel, therefore uneven expansion at elevated temperatures is not a concern. The copper subassembly is not needed for criticality control. For more details on the selection of the copper alloy, see HNF-SD-SNF-ER-019, *Evaluation of Copper for Divider Subassembly in MCO Mark IA and Mark IV Scrap Fuel Baskets* (Graves 1997b).

Other metals that are used in smaller amounts are the soft metal coatings used on the Inconel and stainless steel seals (i.e., inert metals such as silver that will have no adverse reactions with the stainless steel during the life of the MCO).

3.2.5.2 Material Corrosion Resistance. This assessment of chemical and galvanic reactions between the MCO and its environments is divided into three subsections that correspond to the three stages or time periods of operation. The first stage occurs when the MCO is submerged in the K Basins or afterward when it still contains liquid water. The second stage covers the process of water removal and cold vacuum drying. The third stage, without liquid, extends through long-term interim storage.

Assessments of chemical reactions with the environments internal and external to the MCO are predicated on effective control of cleanness during fabrication, handling, and storage of MCO components before and during use. Standards such as ASTM A 380-96, *Cleaning, Descaling, and Passivation of Stainless Steel Parts, Equipment, and Systems* (ASTM 1996a), and ASME NQA-1, *Quality Assurance Requirements for Nuclear Facility Applications* (ASME 1994), are followed for cleanness control.

The MCO is fabricated using welded construction without post-welding heat treatment, so residual stresses in and adjacent to the welds may reach yield strength levels. In an aggressive environment, the MCO could be susceptible to stress corrosion cracking near the welds. The

selection of a low carbon stainless steel was made to minimize the potential for stress corrosion cracking. Other conditions that also minimize the potential for stress corrosion cracking are discussed below.

3.2.5.2.1 Multi-Canister Overpack Containing Liquid Water. The MCO is immersed in or filled with liquid water for less than 2 days, which is too short for significant corrosion in benign environments.

A properly fabricated and cleaned 304L stainless steel MCO rapidly develops a passive corrosion protective oxide film in air. For submerged service, the film needs oxygen for damage repair. However, this protection is typically retained in natural waters, whether hot or cold, even those with relatively high pollution levels (Butler and Ison 1966). According to the MCO Performance Specification (Goldmann 2000b), the conductivity of water in the K Basins ranges from 1 $\mu\text{S}/\text{cm}$ to 5 $\mu\text{S}/\text{cm}$, which is only slightly higher than that of good quality distilled water but significantly lower than that of excellent quality raw water (ASTM 1996b).

The protective oxide film ensures a low rate of corrosion that precludes any damage to the MCO for many years. Common sources of corrosion resistance information do not list typical values for this low uniform corrosion rate. WHC-SD-W236A-TRP-001, *Multi-Function Waste Tank Facility Corrosion Test Report (Phase I)* (Carlos 1993), reports one example of a low corrosion rate in 304L stainless steel such that the predicted corrosion in 75 years would be 0.038 mm (1.5×10^{-3} in.). A design corrosion allowance is not required at this level of corrosion. The Nitronic 60 alloy, also an austenitic stainless steel, is expected to exhibit corrosion resistance similar to that of 304L stainless steel.

The MCO is susceptible to localized corrosion processes (e.g., pitting, crevice corrosion, or stress corrosion cracking) under certain water conditions. The most damaging condition for stainless steels is high concentration of the chloride ion. Chloride ion content in the K Basins is below the detection limit (Goldmann 1998), which is 0.083 p/M by weight, and well below that needed for protection against attack in fully submerged service. The fluoride ion is typically of concern for localized corrosion. The fluoride ion content of K Basin water is 0.248 p/M (Goldmann 1998). High-quality water typically used for mixing cleaning solutions, rinsing, and flushing of nuclear components would contain less than 1 p/M fluoride ion (ASME 1994). Therefore, the fluoride ion will not cause localized corrosion during the water-containing stage. High temperature water containing dissolved oxygen can cause stress corrosion cracking of sensitized stainless steel (Sedricks 1992); however the low water temperature precludes this stress corrosion cracking.

Low levels of polychlorinated biphenyls (PCBs) associated with fuel and sludge corrosion products have been identified in sludge samples from the K East Basin, and their detection raises a concern for thermal or radiolytic decomposition that might contaminate the water in the MCO with chlorine and thereby produce corrosion damage. The corrosion rate of 316 stainless steel in water saturated with chlorine at room temperature is 0.008 mm/yr (ASM 1987, pages 1170-1174), a value that would be acceptable for the short duration of submerged service. A specific corrosion rate for 304L stainless steel is not available; however, it is not expected to be

significantly different than that for 316 stainless steel. In addition, PCBs decompose slowly, levels are low, and they are not expected because the fuel is cleaned before it is loaded into the MCO.

Iodine is a fission product generated during the irradiation of N Reactor fuel; each MCO will contain about 180 g (0.4 lb) of iodine (Praga 1998a). The iodine in light-water-reactor oxide fuel combines with cesium as cesium iodide (Kohli 1982). This compound can vaporize in oxide fuel and migrate to the fuel-cladding gap by vapor transport along pellet-to-pellet interfaces; however, this behavior is unlikely in N Reactor SNF, which has no fuel pellets or fuel-cladding gap. Iodine (or CsI) could be released as the SNF corrodes. Assuming that the iodine would be distributed uniformly, that the cladding does not exist, that the total uranium surface of the original fuel is exposed to corrosion for 48 hours, and that the corrosion rate of uranium in water is $0.57 \times 10^{-3} \text{ g/cm}^2/\text{h}$ (ASM 1987, p 814) results in an upper bound estimate of 0.7 p/M maximum iodine content in the 500 L (130 gal) of water in the MCO. The corrosion literature does not identify iodine or the iodide ion as a major corrosion contributor for stainless steel; however, the low level determined for iodine would be acceptable even for the more corrosive chloride ion. Therefore, iodine contamination is not a corrosion concern.

Cesium is a fission product in N Reactor SNF, and each MCO will contain about 1.2 kg (2.6 lb) of cesium (Praga 1998a). Experience at the K Basins shows that cesium is the major source of radioactivity. Using the same corrosion rate, surface area, and time for uranium corrosion as applied above for iodine, an estimate of maximum cesium content in the MCO water after 48 hours of corrosion is about 4 p/M. There is no evidence in the literature or in K Basin operational experience that cesium is detrimental to the corrosion resistance of stainless steel.

Other fission products present in N Reactor SNF in very small quantities (Praga 1998a) are dissolved in the water and do not enhance corrosion of stainless steel.

Aluminum is highly resistant to high-purity water (distilled or demineralized) at ambient temperatures, with any slight reaction initially occurring ceasing almost completely within a few days after development of a protective oxide film. After this protective film conditioning period, the amount of metal dissolved by the water becomes negligible (Hollingsworth and Hunsicker 1987). Measured corrosion rates in the K East Basin for either a 5086 or 6061 wrought aluminum are less than $0.5 \text{ } \mu\text{m/yr}$ (0.02 mils/yr); similar corrosion rates are expected for the cast aluminum alloy.

Minerals in water combine with dissolved CO_2 and oxygen and react with copper to form a protective film. In distilled or very soft water, protective films are less likely to form; the corrosion rate may vary from less than $2.5 \text{ } \mu\text{m/yr}$ to $125 \text{ } \mu\text{m/yr}$ (0.1 mil/yr to 5 mil/yr) or more, depending on oxygen and CO_2 content (Polan 1987). Even at the higher corrosion rate, impact to the divider subassembly would be minimal because of the relatively short (far less than 1 year) exposure times.

Water is an electrolytic conductor, so the potential for galvanic corrosion has been examined for several dissimilar metal contact scenarios.

1. **Contact between stainless steel and the aluminum fuel rack** — In many environments, including freshwater, aluminum can be used in contact with stainless steels with slight acceleration of corrosion. Stainless steels are easily polarized cathodically in mild environments, so the corrosion current is small (Hollingsworth and Hunsicker 1987). The ratio of stainless steel to aluminum is very large, yet even with an assumed thousand-fold corrosion rate increase, accelerated corrosion of the aluminum would only be 0.5 mm/yr (20 mil/yr).
2. **Contact between stainless steel and the copper divider subassembly** — Both copper and stainless steel exhibit protective passive oxide layers on their surfaces with passive stainless steel more noble (corrosion-resistant) than copper in the sea water galvanic series.¹⁰ This results in accelerated galvanic corrosion of the copper. Only one scrap basket is loaded into each MCO, so the area ratio of stainless steel to copper is very large, increasing the copper corrosion rate.¹¹ The driving force for this galvanic corrosion will be appreciably reduced by the much lower conductivity of the K Basin water (versus sea water) and the relatively small spread between copper and stainless steel in the galvanic series.
3. **Contact with Zircaloy-2 fuel cladding** — The zirconium-based cladding and the stainless steel alloys each exhibit passive oxide layers on their surfaces. Both exhibit similar galvanic corrosion potentials in seawater (ASM 1987, p 717-718), thus there should be no accelerated galvanic corrosion for this alloy combination. The cladding's passive oxide film is much more noble than either the aluminum or the copper oxide film. Therefore, any accelerated galvanic corrosion is expected to be of aluminum and copper.
4. **Contact with aluminum SPR fuel cladding**¹² — Accelerated corrosion of the aluminum fuel cladding with stainless steel would be small (see item 1). The protective oxide coating on the aluminum is less noble than the copper oxide coating. In this case, accelerated corrosion of the aluminum cladding would occur.
5. **Contact with uranium fuel** — The uranium is actively corroding with a nonprotective oxide layer resulting in no accelerated corrosion of either the stainless steel or copper. A previous galvanic couple test (Weirick 1987) of uranium with aluminum

¹⁰Copper can accelerate corrosion of active stainless steel, particularly in highly chlorinated water. However, as chloride levels of the K Basin water are very low and stainless steel is easily passivated, this scenario is not expected.

¹¹Filling an MCO with more than one scrap basket would reduce the stainless steel-to-copper area ratio, thus reducing the magnitude of the copper corrosion rate increase.

¹²8001 aluminum alloy, which has the following chemical additions: 0.9 to 1.3% nickel, 0.45 to 0.7% iron, 0.17% silicon and 0.15% copper (ASM 1990, p. 1456).

in 100% relative humidity showed only a slight corrosive attack for both the aluminum and the uranium.

6. Contact between stainless steel and silver — Silver is only slightly more cathodic than stainless steel, thus galvanic attack of the steel will be negligible.

Even with accelerated galvanic corrosion, the short duration of MCO immersion in water is insufficient for significant corrosion of the construction materials.

3.2.5.2.2 Multi-Canister Overpack during Removal of Liquid Water. Less than 48 hours is needed to remove water from the MCO and establish a low internal water vapor pressure (Goldmann 1998). This period is too short for significant corrosion of stainless steel, aluminum, or copper in the benign environment.

The vacuum drying operation includes monitoring of pressure increases near the end of the process to ensure that acceptable water vapor partial pressure has been established. The water vapor pressure (<0.5 torr) prevents condensation inside the MCO. The single wet/dry cycle precludes significant buildup of chloride ions to levels that would cause localized corrosion. Once the liquid water is removed and condensation is precluded, liquid (galvanic) corrosion processes cease.

If liquid water is trapped in locations such as cracks or crevices in the fuel elements, a complex flow path might produce slow evaporation kinetics that could allow water to remain after drying and sealing of the MCO. This liquid would slowly evaporate into the gas space during the staging period. If enough water vapor were produced to exceed the saturation pressure, condensation on the slightly cooler wall of the MCO could occur. The question then becomes whether this condensate could dissolve sufficient chloride ion from the previously dried walls to exceed the threshold for pitting or crevice corrosion of 304L stainless steel. Evaporation of the low-chloride K Basin water should not create a chloride concentration problem that would enhance pitting or stress corrosion cracking under dry or fully immersed conditions. However, the situation with a relatively small amount of condensate is unclear (Blackburn 1995). Long-term test programs did not reveal significant pitting of 304L in 15-year exposures in a marine environment with much higher chloride concentrations (Davison et al. 1987, Southwell et al. 1976).

Water also is removed from the annulus between the shipping cask and the MCO. Any moisture remaining in the annulus will not produce corrosion damage on the exterior of the MCO during the short time required for shipment to the CSB and removal from the shipping cask.

3.2.5.2.3 Multi-Canister Overpack after Removal of Liquid Water. Following cold vacuum drying, four gases may exist within the MCO in addition to inert gases:

- Hydrogen gas generated by reactions of the uranium fuel with water vapor or radiolysis of chemically bound water

- Chlorine gas produced by thermal or radiolytic decomposition of PCBs, detected in low levels in K Basin sludge samples
- Iodine gas that could be present because of the thermal environment and fuel corrosion
- Oxygen gas generated by the radiolysis of water.

Hydrogen will not reduce the chromium oxide passive layer on the stainless steel although it may reduce the iron oxide that may co-exist in mixed oxide layers (Adams 1983). Effects of gaseous hydrogen on the mechanical properties of 304L stainless steel are discussed in detail in the next section. Aluminum is considered to be resistant to hydrogen at temperatures approaching aluminum's melting point of 660 °C (1,220 °F) (Berry 1971). Dry hydrogen gas is not detrimental to aluminum alloys; however, with the addition of water vapor, subcritical crack growth increases dramatically. It is more common to form a multitude of near-surface voids that coalesce to produce a large blister (Craig 1987). A common form of hydrogen damage in copper is known as steam embrittlement and is observed only when copper contains oxygen. Deoxidized coppers with high residual deoxidizer contents (such as the C12200 used in the scrap baskets) are not considered susceptible to hydrogen embrittlement because the oxygen is tied up in complex oxides that do not react appreciably with hydrogen (Polan 1987).

Dry chlorine is compatible with stainless steels at normal pressures, but chlorine gas saturated with water vapor at ambient temperature is extremely corrosive to these alloys (Brown et al. 1947). In chlorine gas, aluminum is usable up to 120 °C (250 °F), and moisture at room temperature increases attack. A maximum-use temperature of 205 °C (400 °F) is suggested for copper in dry chlorine (Liening 1987); water vapor at room temperature accelerates attack of copper. However, PCBs in the K Basin canister sludge are identified at low levels, so they should not be present within the MCO at any measurable level. This, coupled with cleaning of the fuel before MCO loading and gas purging before storage, further reduces the possibility of chlorine gas corrosion problems.

If the total 180 g (0.4 lb) of iodine contained in the fuel in an MCO (Praga 1998a) were released, the iodine partial pressure would be 26 torr at atmospheric temperature. In actuality, only a small fraction of the iodine would be expected to be released. Assuming the iodine partial pressure to be about the same as that for chlorine, the fact that iodine is less aggressive means that corrosion damage of the stainless steel, aluminum, or copper is very unlikely.

The passive film on 304L stainless steel that protects against liquid corrosion also protects against gaseous oxidation by impurities (e.g., oxygen or water vapor) in the inert gas environment established in the MCO (Adams 1983). Oxidation of stainless steel only becomes obvious at temperatures above about 400 °C (750 °F) (ASM 1987, pages 351-353). Oxygen gas has no effect on aluminum as it aids in the formation of a protective oxide coating (Chawla and Gupta 1993). When copper is used at high temperatures in oxygen, scaling results. Below 100 °C (212 °F), the oxide film increases in thickness logarithmically with time (Polan 1987). At medium temperatures, the scaling rate increases following the parabolic law.

With the liquid removed from the MCO, galvanic corrosion is no longer possible. However, direct contact between the fuel and baskets could lead to liquid metal embrittlement of the stainless steel, aluminum, or copper alloys by fission products or actinides (such as plutonium). Cesium and tin are the low melting point ($<205\text{ }^{\circ}\text{C}$ [$<400\text{ }^{\circ}\text{F}$]) fission products generated in the greatest amounts. However, the tin and cesium levels in the fuel elements are small, so the content in a contact area would be far too small for significant damage to occur to either the stainless steel, aluminum, or copper alloys. Solid metal embrittlement has been observed only in those metal couples in which liquid metal embrittlement occurs (ASM 1987, page 185). A literature search did not reveal any data on solid metal embrittlement of stainless steel by cesium, tin, or any of the actinide metals.

If eutectic liquid could form because localized fuel reactions produced small regions of very high temperature, attack on stainless steel could be severe. Estimates of the lowest temperatures required for liquid eutectic formations obtained from binary-phase diagrams are $646\text{ }^{\circ}\text{C}$ ($1,195\text{ }^{\circ}\text{F}$) for aluminum-uranium and $725\text{ }^{\circ}\text{C}$ ($1,336\text{ }^{\circ}\text{F}$) for iron-uranium. In addition, the melting point of aluminum is $660\text{ }^{\circ}\text{C}$ ($1,220\text{ }^{\circ}\text{F}$) (ASM 1973). Other eutectic temperatures for the binary systems of interest among uranium, zirconium, and copper are higher. Calculations for temperatures within the MCO during interim storage have resulted in a maximum value of $153\text{ }^{\circ}\text{C}$ ($307\text{ }^{\circ}\text{F}$) (Reilly 1998).

The environment at the exterior of the MCO will contain both water vapor and oxygen (either as air or as an impurity in inert gas). The passive oxide layer on the stainless steel will prevent significant reaction with these gases. Internal heat generation in the MCO acts to prevent moisture condensation on the exterior of the MCO. The following engineered and administrative features protect against accidental intrusion of water into the storage tubes at the CSB:

- A dry roof that does not collect water
- Absence of sprinklers for fire protection
- Prohibition against washing the deck
- Seals on the CSB storage tube plugs.

3.2.5.2.4 Hydrogen Effects on the Mechanical Properties of Stainless Steel. Hydrogen gas is a principal contributor to the internal pressure in the MCO. The allowable gas amounts defined in HNF-SD-SNF-OCD-001, *Spent Nuclear Fuel Project Product Specification* (Pajunen 2000), show the sum of the total water and hydrogen plus the contingency put an upper limit on the hydrogen pressure of 124 lb/in^2 absolute at a temperature of $200\text{ }^{\circ}\text{C}$ ($392\text{ }^{\circ}\text{F}$). An extensive compilation of the effects of hydrogen on the mechanical properties of 304L stainless steel is provided in DP-1643, *Hydrogen Compatibility Handbook for Stainless Steels* (Caskey 1983). Much of the experimental information was obtained for a pressure of $10,000\text{ lb/in}^2$, either as an external hydrogen environment during the test or as a pressure for charging hydrogen internally into the steel at elevated temperatures. Only in the case of tensile ductility are sufficient data available at lower pressures to determine effects at the MCO pressure.

Following Sievert's Law, the concentration of hydrogen just inside the alloy surface is directly proportional to the square root of the hydrogen pressure (Caskey 1985, page 830). The

MCO pressures will be much less than those employed in Caskey's tests (1983), hence, so will the hydrogen concentrations. The extent of hydrogen diffusion into the steel for two MCO CSB conditions and two Caskey hydrogen charging conditions was calculated by combining Fick's second law and the diffusivity temperature-dependant Arrhenius equation (ASM 1985, pages 28-65 and 28-66) to yield

$$x \approx \{D_0 t [\exp(-Q/RT)]\}^{1/2}$$

where

x = hydrogen diffusion distance (cm)

D_0 = diffusivity constant (2×10^{-3} cm²/s for 304L [Caskey 1985, page 828])

t = time (s)

Q = activation energy (50 kJ/mol for 304L [Caskey 1985, page 828])

R = universal gas constant (8.314 J/mole·K)

T = temperature (degrees kelvin [$K = ^\circ C + 273$]).

The Caskey conditions calculated were (1) 197 °C for 1,449 days (Caskey 1983, page 81) and (2) 347 °C for 3 weeks (Caskey 1983, page 83). For both of these conditions, the diffusion distance was equal to or greater than the original diameter of the tensile specimen, thus full penetration of the hydrogen was achieved. The MCO conditions calculated were (1) 132 °C (maximum MCO shell temperature in CSB) for 40 years and (2) 46 °C (maximum air temperature within CSB storage tubes) for 50 years. The calculated diffusion distances for the MCO conditions were 0.373 in. and 0.056 in., respectively. These distances correspond to 75% and 11% of the MCO shell wall thickness, thus limiting the hydrogen to the inner surface of the MCO. Consequently, experimental results from Caskey's hydrogen effect tests (Caskey 1983) conservatively bound effects for the MCO. Based upon these calculations, thinner components, less than 0.35 in. thick, within the MCO (i.e., the rupture disk) will achieve full hydrogen penetration.

The following summary of information from DP-1643 (Caskey 1983) contains parenthetical reference to specific figures or pages of that document.

- Ductility

The most commonly used index of hydrogen damage in stainless steels has been the change in reduction-of-area as measured for a fractured tensile specimen. The reduction-of-area is a measure of plasticity calculated from the original cross-sectional area (A_0) and the final cross-sectional area at the fracture (A_f).

$$RA = 100 (A_o - A_f)/A_o.$$

Another measure of ductility that is used extensively in DP-1643 (Caskey 1983) is plastic strain to failure (E_f).

$$E_f = \ln (A_o/A_f).$$

High hydrogen pressure can reduce reduction-of-area from a starting value of about 80% to a value of about 22% at a temperature of about -53 °C (-63 °F), which corresponds to a minimum in reduction-of-area (these values were calculated by converting Caskey's E_f values (Caskey 1983) to reduction-of-area values using the previous equations). However, for a hydrogen pressure of about 450 lb/in², the reduction-of-area would only be reduced to about 61% at about 22 °C (72 °F). This level of reduction-of-area is typically more than adequate to ensure ductile structural behavior in engineering components. At a service temperature of 200 °C (392 °F), the reduction-of-area value would be even higher than 61% (Caskey 1983, Figures 12 and 13, pages 81, 83, 86).

- Yield strength

High-pressure hydrogen produces small increases of about 0% to 28% in the yield strength of 304L stainless steel (Caskey 1983, pages 24, 31, 81, 82, 83).

- Tensile strength

High-pressure hydrogen typically produces small decreases of about 10% to 15% in the tensile strength (Caskey 1983, pages 31, 81, 82). These small reductions do not influence design allowable stress intensity because this parameter is governed by yield strength for conditions applicable to MCO storage.

- Notch strength

Stainless steels like 304L in conventional tensile tests are typically strengthened by notches in the absence of hydrogen (the opposite behavior indicates susceptibility to brittle fracture at stress concentrations). High-pressure hydrogen produces a reduction of less than 20% in the notch strength (Caskey 1983, pages 47, 88, 89).

- Elastic/plastic fracture toughness

High-pressure hydrogen produces reduction in the J-integral at maximum load of about 30% and in the tearing modulus of about 20% (Caskey 1983, pages 84, 85). These changes are much too small to be of practical engineering significance for the MCO.

- Static crack growth

Slow crack growth under static loads did not occur in fracture mechanics tests of thin specimens of 304L stainless steel. Crack growth did occur in notched specimens loaded to 85% of the notch tensile strength (Caskey 1983, pages 50, 51, 52). As the MCO design does not allow loads to reach this critical level, static crack growth is of no concern.

- Impact energy

Impact tests of a dynamic tear test specimen showed only a small decrease in absorbed energy for tests in hydrogen at room temperature. Even at -196°C (-321°F), absorbed energy values did not indicate brittle fracture (Caskey 1983, pages 81, 83).

- Stress state

Burst testing of disks produces a biaxial stress state in the test specimen. Tests using hydrogen as the pressurizing gas show little change in burst pressure relative to helium tests for solution-annealed 304 stainless steel, but a reduction of about 45% in burst pressure for samples that were sensitized or welded (Caskey 1983, page 46; Fidelle et al. 1974).

With the exception of yield and tensile strength, the material properties discussed above are not design parameters. The strength values used during the critical analyses are conservative values as required by the applicable codes and take into account slight property variations. The hydrogen effects on those material properties not used in the design calculations (i.e., notch toughness) show no significant loss in strength, ductility, or resistance to crack propagation that would adversely affect the design, analysis, or structural performance of the MCO.

Hydrogen Compatibility Handbook for Stainless Steels (Caskey 1983) was used as a source because it is more definitive than any other sources found in the original literature search. The Handbook is a compilation of 15 years of research at the Savannah River Laboratories, supplemented by other available data. A later literature search did not yield any additional data on hydrogen embrittlement at low pressures (~ 1 MPa). Data found were similar to Caskey's data and for pressures greater than 30 MPa.

3.2.6 Lifting Devices

Various devices will be used on the MCO and its components during loading, fuel conditioning, and placement into storage. A general description of these follows.

- Device to move and place the empty MCO canister — A pipe tong device has been selected.

- Device to place the MCO shield plug — Design is underway, and device will attach to set screw holes in the plug upper surface.
- Device to place and install the MCO lifting and locking ring — Design is under way, and device will both lift the ring onto the MCO canister and torque the set screws.
- Device for lifting the fuel baskets — The device is a grapple that attaches to a groove internal to the basket's center pipe uses three balls locked in a radial position providing the axial load path (together with the pipe groove) for lifting the baskets. This device has been designed, analyzed, and tested to the limits stated in ANSI N14.6-1993, *For Radioactive Materials — Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More* (ANSI 1993), to show acceptance for these lifts.
- Device for lifting and moving the loaded MCO — The MHM has been designed under the direction of the CSB subproject. The MHM uses the collar or cover cap lifting ring for attachment.

3.2.7 Fuel Elements

At no time during retrieval, packaging, transportation, or conditioning is the fuel cladding relied upon for confinement of radionuclides. Fuel movements in the fuel segregation campaign, and more recently during the fuel characterization work, have shown that even the most visibly damaged fuel can be moved and handled. For criticality analyses, it is assumed that the cladding remains mixed with the fuel.

3.3 SUPPLEMENTAL DATA

3.3.1 Deflagration

The effects of a deflagration occurring at atmospheric pressure either inside or outside of the MCO pressure boundary have been addressed (Shrivastava 1996, Shrivastava 1997). The analyses show that the MCO will meet the ASME Boiler and Pressure Vessel Code, service level D requirements (ASME 1998) for internal pressures expected from a hydrogen deflagration event starting at atmospheric pressure. The maximum possible deflagration pressure occurs from a hydrogen-oxygen stoichiometric mix initially at 1 atm pressure with a reflected shock wave of 37. This shock wave could produce a pressure of 544 lb/in². This shock wave pressure has been predicted to be more realistically limited to a value of 13 atm. This would yield a pressure wave of 190 lb/in², which is a factor of 2.4 below the MCO design pressure of 450 lb/in² with cover cap attached.

The MCO pressure boundary vessel, when exposed to an internal deflagration, has been shown to meet the service level D service limits of the ASME Boiler and Pressure Vessel Code (ASME 1998) with pressures up to 935 lb/in². This is ample margin on a very conservatively predicted deflagration pressure.

The MCO pressure boundary vessel, when conservatively addressed for an external deflagration event acting on a vessel not supported internally, shows a pressure limit of 354 lb/in². Should one consider the internal vessel supports of the basket bottom plates, the allowable pressures could be shown to be much higher.

3.3.2 Multi-Canister Overpack Cover Plates

Analyses (Abatt 1997) of the cover plates when impacted by the cask lid, if it were accidentally dropped at heights up to 5 ft above the MCO, show the plates adequately protect the port appliances. The cover plates limit displacements toward the appliances to 0.34 in. (Abatt 1997). The appliances are designed with at least this much clearance, so they will suffer no impact or resultant forces.

3.4 COMPUTER CODES

The SNF primary confinement boundary structure, the MCO, was analyzed for structural strength and dynamic characteristics using the finite element codes described in the following paragraphs. The computer codes are those commonly used in the nuclear industry and particularly at the Hanford Site. The codes all have been validated and inputs verified for quality answers by user groups onsite and by the design agent.

ANSYS 5.0A (Swanson 1992) is a finite element program that has acceptance as a linear elastic design tool for mechanical and civil engineers. It is known for its early development and use as a robust pre- and postprocessor. ANSYS is being expanded to nonlinear analysis, but it is still viewed as a standard for linear elastic structural analysis in the nuclear industry. ANSYS has been used extensively in the structural static analyses of the MCO and its components. The code has the ability to develop stresses related to thermal growths and restraints in structures.

For dynamic solutions, in particular the cask-MCO drop scenarios, the K Basin transportation cask, MCO, and baskets were analyzed and modeled with a second generation, nonlinear, finite element program called ABAQUS (ABAQUS 1995b). ABAQUS uses finite element formulation for the stress and strain domain and finite difference formulations for the dynamic time domain. There are two versions of ABAQUS: ABAQUS/Standard, the implicit formulation, which is best for static nonlinear solutions; and ABAQUS/Explicit, which is best for dynamic problems. In particular, ABAQUS/Explicit 5.4 was used for this analysis effort (ABAQUS 1995a). ABAQUS/Explicit 5.4 uses a lumped-mass formulation with explicit central difference integration for the time domain solution of displacements, velocities, and accelerations. Each degree of freedom is idealized as an independent lumped mass with an initial position and

velocity. External forces are applied to generate accelerations. Employing very small time steps (typically 0.5 microsecond), new positions and velocities are calculated, then finite element reaction forces are calculated for each individual element, and contact surface conditions are checked. The new updated forces produce new accelerations and the process repeats. ABAQUS also allows nonlinear modeling of materials and material failure. The nonlinear material properties are defined versus strain. The ABAQUS finite element formulation uses numerical integration to construct element stiffness matrices that relate internal element reaction forces produced by element boundary displacements. The repeated numerical integration allows the material properties to be updated based on the strain history at each element integration point through the element volume.

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Figure 3-1. Multi-Canister Overpack and Cask Package Sketch.

Figure 3-2. Axisymmetric Model with Boundary Conditions, Upper Section.

Figure 3-3. Axisymmetric Model with Coupled Nodes.

Figure 3-4. Axisymmetric Model with Boundary Conditions, Lower Section.

Figure 3-5. Multi-Canister Overpack Half-Symmetry
Assembly Without Cover Cap.

Figure 3-6. Multi-Canister Overpack Half-Symmetry
Assembly With Modified Cover Cap.

Figure 3-7. One Hundred Eighty Degree Sector of Mark IA Fuel Basket Structural Components.

Figure 3-8. Support Rod Finite Element Model.

Figure 3-9. Support Rod Force Deflection Curve (Single Rod).

Figure 3-10. Support Rod Deformed Shape.

Figure 3-11. Sixty Degree Sector Model of Mark IA Basket with Holes.

Figure 3-12. Illustration of Support Rod Load Eccentricity.

Figure 3-13. Mark IA Support Rod Cross Section.

Figure 3-14. One Hundred Eighty Degree Model of Mark IA Basket Used for Horizontal Drop Analyses.

Figure 3-15. Elastic Stress Intensity Contours, 35 g Drop Loading,
Detailed Hole Model.

Figure 3-16. Elastic/Plastic Stress Intensity Contours,
35 g Vertical Drop.

ANSYS 5.3
 FEB 18 1998
 17:41:28
 PLOT NO. 1
 NODAL SOLUTION
 STEP=1
 SUB =10
 TIME=1
 SINT (AVG)
 TOP
 DMX =.033332
 SMN =2987
 SMX =71525
 A =6795
 B =14410
 C =22025
 D =29641
 E =37256
 F =44872
 G =52487
 H =60102
 I =67718

Figure 3-17. Plastic Strain Contours, 35 g Vertical Drop.

Figure 3-18. Bottom Plate Distortion Illustration for Horizontal Drop Modeling.

Figure 3-19. 101 g Horizontal Drop Stress Intensity Contours,
Inelastic Analysis.

Figure 3-20. Sixty Degree Sector Finite Element Model of the Mark IV Storage Basket.

Figure 3-21. Mark IV Support Rod Cross Section.

INTERFACE - SUPPORT ROD TO BASE PLATE

Figure 3-22. Shield Plug Interface Component Geometry.

Figure 3-23. Cover Cap With Gripping Shoe Configuration.

Figure 3-24. Thread Detail Geometry.

Pitch = 0.25 in.
 h_o = 0.1410 in.
 G = 0.06115 in.

h_s = 0.1660 in.
 F = 0.04080 in.

**This document was too large to scan
as a single document. It has
been divided into smaller sections.**

Section 2 of 3

Document Information			
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4.0 THERMAL EVALUATION

This chapter presents the thermal-hydraulic evaluations used to establish the safety basis for the multi-canister overpack (MCO) and its spent nuclear fuel (SNF) payloads under normal conditions of operation. The safety basis for off-normal and accident conditions is addressed by the various facility safety analysis reports and by the safety analysis report for packaging for the transportation cask. This division in reporting complies with the format guidance of U.S. Nuclear Regulatory Commission Regulatory Guide 3.61, *Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask* (NRC 1989a), wherein the thermal evaluation for a topical report presents the normal conditions of storage.

Other chapters of this topical report provide supporting data. Chapter 1.0 presents the general description of the MCO, the Mark IV and Mark IA fuel baskets, the Mark IV and Mark IA scrap baskets, and the associated design and assembly drawings. The material and dimensional data provided in Chapter 1.0 are used in the thermal models of the MCO assembly. Chapter 2.0 provides the physical, chemical, and radiological description of the SNF fuel types to be handled, the design loadings for the MCO, the environmental design conditions, and the safety protection systems inherent with the MCO design. These data are used as an aid in establishing the heat transfer characteristics of the SNF fuel types, the boundary conditions for the thermal analyses, and the design thermal loading for the MCO. Chapter 8.0 defines the MCO configuration for each processing step. This information is used in setting boundary conditions for the thermal analyses. Finally, Chapter 12.0 defines key parameters used as sources for facility-specific technical safety requirements to ensure system safety. These data support the determination of the thermal acceptance criteria.

4.1 DISCUSSION AND RESULTS SUMMARY

From loading at the K Basins to placement in interim storage, the interior of the MCO sees three distinct thermal-hydraulic conditions: flooded with water, vacuum or near vacuum conditions, and dried and backfilled with helium gas. While combinations of these three distinct conditions exist as an MCO passes from one process step to the next, the safety basis of the MCO is principally established using its steady-state and transient thermal performance under each of these distinct operating conditions.

Water-flooded thermal conditions exist during loading at the K Basins, transportation between the K Basins and the Cold Vacuum Drying Facility (CVDF), and during the initial processing steps at the CVDF. The safety analysis under water-flooded conditions consists primarily of ensuring the system will contain any potential pressure buildup because temperature excursions are virtually impossible so long as the SNF assemblies remain covered with water.

A combination of water-flooded, vacuum, and dry thermal conditions exist during the various phases of the cold vacuum drying process. The safety analysis for cold vacuum drying consists of two parts. The first part addresses the cold vacuum drying process steps to ensure

that the SNF temperatures remain within the safety limits when exposed to the elevated temperatures, water vapor, and reduced heat transfer environment inherent with the vacuum drying process. The second part of the cold vacuum drying safety evaluation addresses the process steps required to ensure that the water and other reactants remaining within the MCO following cold vacuum drying are below set maximum quantities. Ensuring that the reactant quantities are limited will prevent excessive temperature and pressure excursions within a sealed MCO during the subsequent dry transportation and interim storage periods.

Dry thermal conditions within the MCO exist following completion of the cold vacuum drying process steps, during transportation between the CVDF and the Canister Storage Building (CSB), and during handling and interim storage at the CSB. Because it is recognized that a finite amount of water may exist following the cold vacuum drying process as either free or chemically bound water, the terminology 'dry' is used to refer to the prevalent environmental condition within the MCO. The safety analysis under these conditions consists of establishing the maximum temperatures and pressures that may occur within the MCO during transportation to the CSB, handling within the MCO handling machine (MHM), and during interim storage in the CSB given the amounts of reactants still remaining within the MCO.

Table 4-1 summarizes the results of the thermal-hydraulic safety analysis for the MCO assembly. The table addresses the type of analysis used to establish the safety basis, provides a comparison of key MCO thermal parameters with acceptance criteria, and indicates whether a technical safety requirement is necessary to ensure the safety basis. The key thermal parameters are selected to provide a means of tracking the overall thermal performance of the system. The 'allowable' temperature represents the maximum temperature permitted for that component, based on a code or manufacturer's restriction, while the 'design' temperature represents the temperature used for design purposes by nonthermal analyses.

As seen from the table, the specific thermal parameters that are tracked for the MCO are the MCO wall temperature, the temperature difference between the shield plug and the MCO wall, the maximum SNF temperature, and the MCO gas pressure. The thermal-hydraulic analyses presented herein demonstrate that all SNF processing steps comply with the thermal acceptance criteria for these thermal parameters and that substantial thermal margins exist.

4.1.1 General Analysis and Key Parameter List

Generally, a safety basis analysis would address the range of expected environmental and process conditions and demonstrate a safe, stable, and secure operation under all processing steps. This demonstration would involve maintaining a passive energy balance wherein the energy gains to the system equal the energy losses from the system without reliance on active systems or operator intervention. However, the potential worst-case composition of the MCO's SNF payload, the exponential relationship between temperature and chemical corrosion heat, and the thermal environment imposed by some of the process steps can combine to create inherently unstable operating conditions. Therefore, in some cases the thermal evaluations for the MCO had to address the transient thermal behavior of the system to establish the safety basis.

Table 4-1. Summary of Results for the Thermal-Hydraulic Safety Analyses of the Multi-Canister Overpack. (2 sheets)

	Limits		Flooded transport ^a	Cold vacuum drying ^b	Dry transport ^c	CSB handling ^d	CSB storage ^e
	Allowable	Design					
Type of analysis conducted for safety basis							
Steady-state analysis	--	--	No	No	Yes	Yes	Yes
Transient analysis	--	--	Yes	Yes	Yes	Yes	Yes
Maximum component temperature (°C)							
MCO wall	149	132 ^f	36	50	76	132	108
Shield plug - Average - Differential	149 <100 ^g	na <100 ^g	33 3	56 35	45 31	111 25	107 5
Rupture disk	132	132	33	na	na	na	na
Maximum fuel	600 ^h	600 ^h	39	92	94	153	131
Maximum fuel basket ⁱ	149	149	39	89	91	148	128
Average MCO gas	--	--	33	51	78	142	122
Maximum pressure (lb/in ² gauge)							
MCO pressure	150 or 450 ^j	150 or 450 ^j	88	0 - 11	32	33	61.7

Table 4-1. Summary of Results for the Thermal-Hydraulic Safety Analyses of the Multi-Canister Overpack. (2 sheets)

	Limits		Flooded transport ^a	Cold vacuum drying ^b	Dry transport ^c	CSB handling ^d	CSB storage ^e
	Allowable	Design					
Technical safety requirements							
TSR required?	-	--	Yes ^a	Yes ^b	No	No	No

^aTemperatures and pressures based on 24-hour shipping window. Safety action required to limit transport time to <24 hours from the time the cask lid is sealed without instituting cask venting. Shipping in ambient temperatures below -18 °C (0 °F) to be prohibited for all MCOs generating less than 200 W of decay heat.

^bTemperatures assume an extended time under vacuum (i.e., 48 hours). Time under vacuum is limited to 8 hours on the first cycle and 4 hours on subsequent cycles; pressure rebound test is required before the MCO is released for transport to CSB (Irwin, J. J., 1999, *Spent Nuclear Fuel Project Cold Vacuum Drying Facility Operations Manual*, SNF-2356, Rev. 1, Fluor Daniel Hanford, Incorporated, Richland, Washington).

^cTemperatures and pressures based on 48-hour shipping window.

^dThe analysis demonstrates that a minimum of 160 hours of operation in the MCO handling machine without forced convection is permissible prior to exceeding the 132 °C design temperature limit for the MCO wall.

^eBased on steady-state operation in CSB vault with design day ambient temperature.

^f≤132 °C according to Goldmann L. H., 2000a, *Multi-Canister Overpack Design Report*, HNF-SD-SNF-DR-003, Rev. 3, Fluor Hanford, Incorporated, Richland, Washington.

^gDifferential temperature between shield plug and MCO shell limited to ≤100 °C according to HNF-SD-SNF-DR-003.

^h≤600 °C according to Table 12-1.

ⁱEstimated temperatures are based on SINDA and GOTH_SNF analysis. The maximum fuel basket temperature will typically occur at the center post with the temperature differential from the peak fuel temperature ranging from nearly 0 °C for water-filled conditions to 3 °C for helium-filled, steady-state conditions. Under transient conditions, such as CSB handling, the temperature differential will increase because of thermal mass effects. See also Section 2.2.6.2.1 of this report.

^jPressure limit increases to 450 lb/in² gauge after the cover cap is welded to the MCO. This will occur after arrival at the MCO and before interim storage.

CSB = Canister Storage Building.

MCO = multi-canister overpack.

TSR = technical safety requirement.

For the purposes of this work, a safe, stable, and secure operating condition is defined as either (1) an energy-balanced, steady-state condition wherein the energy gains to the system equal the energy losses from the system and the system is within its temperature and pressure limits, or (2) a transient operating condition wherein energy gains may temporarily exceed energy losses, but for which administrative controls are implemented to ensure the system can be placed into a stable, steady-state condition before the system temperature and pressure limits are exceeded.

The thermal-hydraulic safety analyses presented in this chapter are based on a combination of bounding assumptions for the radiolytic and chemical heat generation, conservative heat transfer assumptions, and application of technical safety requirements governing the loading of the MCO, allowable time to accomplish individual process steps, process performance criteria that must be met before subsequent process steps can be started, and specified recovery steps to be taken should off-normal conditions arise. The combination of all of these elements ensures that the MCO assembly and its payload will be maintained in a safe, stable, and secure operating condition under the various process steps.

The thermal evaluation for safety presented within this chapter addresses the following elements:

- A description of the thermally significant aspects associated with each process step, from initial fuel retrieval to final placement of sealed MCOs at the CSB facility
- A summary of the project-approved parameters for the critical engineering and physical aspects of the MCO's design and the design of the fuel baskets
- The recommended thermal properties and material specifications for the materials used in the fabrication of the MCO and its fuel baskets and for the SNF payload
- The technical specifications for the vendor-supplied components
- Definition of the potential payloads and the recommended thermal source terms for computing the radiolytic decay and chemical corrosion heat
- The basis for the water content before and after cold vacuum drying
- The basis for computing the pressure rise within the MCO due to hydrogen gas generation and thermal effects
- The predicted thermal results for normal operations at each process step which demonstrate the safety basis for the system.

The key parameters (and their bases) used to assess the thermal-hydraulic safety of the MCO are as follows:

- **MCO wall** — A maximum permissible temperature of 427 °C is set by the ASME *Boiler and Pressure Vessel Code* (Section III, Subsection NB) for type 304L stainless steel (ASME 1998). However, the functional maximum temperature of 149 °C is set by the temperature assumption used in the structural analyses for the design stresses. While MCO wall temperatures in excess of 149 °C could be tolerated, the necessary documented safety analysis associated with a higher temperature does not exist.

In addition to a maximum allowable wall temperature, the MCO wall is limited to an allowable through-wall temperature gradient of 5 °C. This temperature gradient limit is the value used in the structural analyses and was originally established to limit the stresses during the heating and cooling cycles of the hot conditioning process step. Although this process step has been eliminated, the allowable wall gradient of 5 °C is maintained as a bounding design criterion for the remaining MCO process steps even though none of the gradients induced during these steps ever challenge the limit. In fact, based on the maximum design radiolytic heat expected and assuming that all the dissipated heat flows radially out through the MCO wall, the expected wall gradient would be less than 0.1 °C.

- **Shield plug** — Again, a maximum permissible temperature of 427 °C is set by the ASME *Boiler and Pressure Vessel Code* (ASME 1998), while the functional maximum temperature of 149 °C is again set by the temperature assumption used in the structural analyses. The 100 °C limit on the temperature differential between the shield plug and the MCO wall was originally created to limit differential thermal expansion and preserve the integrity of the main MCO seal during the external heating process for hot conditioning. Although it was only under this step that the temperature differential limit was challenged, the limit was retained after the elimination of the hot conditioning step as a structural design basis. As seen from Table 4-1, this limit is never approached for any of the remaining process steps.
- **Rupture disk** — The temperature of 132 °C and pressure of 150 lb/in² gauge for the rupture disk are set by the MCO design.
- **SNF fuel** — The maximum fuel temperature limit of 600 °C is set by the lowest fuel eutectic formation temperature (see Section 12.2.1 for additional discussion).

The various design inputs required for establishing the thermal-hydraulic safety basis for the MCO are extracted from a variety of project documentation. The design inputs that are considered to be critical to the safety basis analysis are listed in Table 4-2, together with their respective sources. The table also presents the location within this topical report where these critical parameter values are discussed. While these values are replicated within this report for the reader's convenience, the project-approved reference points for these critical parameters remain with the underlying referenced documents as presented in the table.

Table 4-2. Critical Parameters for Multi-Canister Overpack Thermal Analysis. (2 sheets)

Design parameter	Source document	Reference within Topical Report
MCO dimensions and weights	HNF-SD-SNF-DR-003, <i>Multi-Canister Overpack Design Report</i> , Rev. 2	Sections 1.2 and 4.1.4 and Tables 3-6 and 4-5
Fuel and scrap basket dimensions and weights	HNF-SD-SNF-DR-003, <i>Multi-Canister Overpack Design Report</i> , Rev. 2	Sections 1.2.2 and 4.1.4 and Table 4-5
Number and position of fuel and scrap baskets within MCO	HNF-SD-SNF-CSER-005, <i>Criticality Safety Evaluation Report for Spent Nuclear Fuel Processing and Storage Facilities</i> , Rev. 3	Section 6.2.3 and Table 4-5
Void volumes in MCO	HNF-SD-SNF-DR-003, <i>Multi-Canister Overpack Design Report</i> , Rev. 2	Table 4-5
SNF dimensions and weights (design basis)	HNF-SD-SNF-TI-009, <i>105-K Basin Material Design Basis Feed Description for Spent Nuclear Fuel Project Facilities</i> , Rev. 2	Table 2-1
SNF condition following retrieval and cleaning	HNF-SD-SNF-OCD-001, <i>Spent Nuclear Fuel Project Product Specification</i> , Rev. 4	Tables 4-4 and 4-5
Mass limit for Mark IV and Mark IA scrap baskets	HNF-SD-SNF-CSER-010, <i>Criticality Safety Evaluation Report for the K Basin Fuel Retrieval Subproject</i> , Rev. 0	Section 6.2.3 and Table 4-5
Fuel surface area per MCO and per basket	HNF-SD-SNF-TI-015, <i>Spent Nuclear Fuel Project Technical Databook</i> , Rev. 6	Table 4-7
Recommended fuel reaction rates and enhancement factors	HNF-SD-SNF-TI-015, <i>Spent Nuclear Fuel Project Technical Databook</i> , Rev. 6	Section 4.2.2.2
Canister particulate mass and water content	HNF-SD-SNF-TI-015, <i>Spent Nuclear Fuel Project Technical Databook</i> , Rev. 6	Tables 4-4 and 4-5
Decay heat per MCO	HNF-SD-SNF-TI-015, <i>Spent Nuclear Fuel Project Technical Databook</i> , Rev. 6	Table 4-6
Process steps within K Basins	HNF-SD-SNF-OCD-001, <i>Spent Nuclear Fuel Project Product Specification</i> , Rev. 4	Section 4.1.3.1
Process steps for transport	HNF-SD-TP-SARP-017, <i>Safety Analysis Report for Packaging, Onsite Multi-Canister Overpack Cask</i> , Rev. 2	Sections 4.1.3.1 and 4.1.3.3 and Table 4-3
Process steps within CVDF	HNF-2356, <i>Spent Nuclear Fuel Project Cold Vacuum Drying Facility Operations Manual</i> , Rev. 1	Section 4.1.3.2

Table 4-2. Critical Parameters for Multi-Canister Overpack Thermal Analysis. (2 sheets)

Design parameter	Source document	Reference within Topical Report
Process steps within CSB	HNF-SD-SNF-OCD-001, <i>Spent Nuclear Fuel Project Product Specification</i> , Rev. 4, and Drawing H-2-123400, <i>Operational Sequence Block Flow Diagram</i>	Section 4.1.3.4

CSB = Canister Storage Building.

CVDF = Cold Vacuum Drying Facility.

MCO = multi-canister overpack.

SNF = spent nuclear fuel.

Generally, the analyses presented herein are based on full compliance with the data from the latest revision of each document listed in Table 4-2. One notable exception is that for Revision 6 of HNF-SD-SNF-TI-015, *Spent Nuclear Fuel Project Technical Databook* (Reilly 1998). Some of the minor changes made as part of Revision 6 were not incorporated into all of the HANSF analytical models because the impact of the changes did not warrant rerunning the analyses. The specific changes that are not consistently addressed are the particulate loading and its attendant water content. While the latest changes to these parameters have resulted in an increase in the total water content within the design and safety basis MCO, the associated effects on the analyses presented herein are only marginal. This results from the limited decomposition of the aluminum hydroxide and the hydrates under all conditions except interim storage. Thus, the slight increase in overall water content brought about by the Revision 6 changes to HNF-SD-SNF-TI-015 (Reilly 1998) will not be the driving factor in any process step, except interim storage. The calculations for the long-term pressure rise during interim storage are based on the latest data in HNF-SD-SNF-TI-015 (Reilly 1998).

4.1.2 Assumed Condition for Fuel and Scrap

Before the SNF assemblies are placed in the fuel baskets, they are mechanically cleaned within their existing storage canisters to remove as much of the accumulated fuel corrosion product as possible. The mechanical loads placed on the assemblies during the cleaning process also aid in assessing the structural integrity of the assemblies before their placement in the intact fuel baskets.

The assumed condition of the fuel assemblies and scrap is principally defined in HNF-SD-SNF-TI-015 (Reilly 1998), HNF-SD-SNF-TI-009, *105-K Basin Material Design Basis Feed Description for Spent Nuclear Fuel Project Facilities, Volume 1, Fuel* (Praga 1998a), and HNF-SD-SNF-OCD-001, *Spent Nuclear Fuel Project Product Specification* (Pajunen 2000). Documents referenced in HNF-SD-SNF-TI-015 (Reilly 1998) provide additional detail as to the condition of the fuel assemblies and scrap. The critical design basis assumed by the thermal analyses is as follows.

- SNF dimensions and weights — The design basis for the fuel assemblies is documented in HNF-SD-SNF-TI-009 (Praga 1998a). Table 2-1 summarizes the key dimensions and weights used in the thermal analyses.
- Adhering particulate and films — While the cleaning accomplished under the initial process step is important for reducing the uncertainty in the makeup of the basket contents, no credit is taken in the safety basis analysis for the expected level of cleaning. This approach avoids the quality assurance requirements that would attend to a specific cleaning requirement. Instead, any cleaning accomplished will serve to increase the safety margin above that predicted by the thermal-hydraulic safety analysis by reducing the amount of particulate adhering to the assemblies and, thus, the amount of chemically bound water
- HNF-1527, *Estimates of Particulate Mass in Multi-Canister Overpacks* (Slougher 1998), and HNF-3247, *Statistical Analysis of Aluminum Trihydroxide Thickness Data* (Jensen 1999), provide the basis for the assumed adhering particulate mass reported in HNF-SD-SNF-TI-015 (Reilly 1998) and for the bounding film thicknesses on the intact fuel assemblies. According to HNF-1527 (Slougher 1998), a bounding uranium surface oxide thickness of 0.4 mils is to be assumed. This film thickness is three times the nominal amount measured in the laboratory on K Basin fuel samples. The thickness of the aluminum trihydroxide film found on some fuel assemblies stored in the K West Basin was measured using eddy current techniques. The instrumentation was calibrated to an accuracy of ± 0.5 mils. HNF-3247 (Jensen 1999) presents the findings of these measurements and provides a statistical analysis for mean and bounding values. Based on HNF-3247 (Jensen 1999), a bounding film thickness of 2 mils is assumed. Assuming a conservatively low film conductivity of 2 W/m-K (for uranium dioxide), a maximum decay heat of 2.87 W per assembly, and an external surface area of 1,281 cm², a 2-mil film thickness would impose virtually no thermal resistance to the radial flow of the decay heat. Given this extremely low thermal resistance, neither an increase in heat flux from chemical corrosion nor a localized thickening of the film will add a substantial temperature increase.

Following cleaning, the assemblies are placed on a sorting table and segregated according to condition and size. The sorting and segregation process is subjectively controlled by the operators according to the following general guidelines.

- Only those portions of a combined fuel assembly (i.e., an outer and an inner element) that will fit in the baseplate socket of a fuel basket are loaded into that type of fuel basket.
- Solo outer or inner elements are either combined to make up a complete fuel assembly (for criticality reasons) or placed in one of the scrap baskets. Mark IA fuel assemblies and Mark IV fuel assemblies will not be mixed in the same MCO.

- Those portions of the fuel assemblies that are greater than 0.25 in. in size but less than approximately 3 in. in length, or are too broken up to stand upright in the intact fuel basket, or will not fit in the baseplate socket because of damage or swelling caused by corrosion, are placed in one of the scrap baskets.
- The scrap basket designs incorporate a fines container at the center of the basket into which the smaller particles of fuel scrap are placed. The volume of the fines container is intentionally set to limit the amount of fine particles allowed within any one scrap basket. A fine particle is defined as those pieces of fuel scrap greater than 0.25 in. in size but less than approximately 1 in. in size. Particles less than 0.25 in. in size are to be handled by a separate process.
- A 0.25-in. screen will be used on the sorting table to assist in segregating fuel pieces according to size.

It is recognized that pieces of SNF smaller than 0.25 in. in size may be unintentionally loaded or created as a result of loading into the basket. This possibility will not violate the general guidelines describe above in that the presence of a few small pieces will not affect the thermal performance of the MCO. In addition, the basket heights will be gauged before shield plug insertion to ensure that the presence of small pieces will not affect the proper assembly of the MCO.

4.1.3 Description of Process Steps

A loaded MCO never exists in a stand-alone configuration. Instead, a loaded MCO is enclosed by either the transportation cask, the MHM, the CSB weld station, or the storage vault within the CSB. Because of this, the thermal-hydraulic performance of the MCO and its payload is dictated by the environment imposed on the MCO by these enclosures and/or by the flow streams introduced into the MCO as part of the various processing steps. To address the thermal-hydraulic performance of the MCO and its payload first requires an understanding of the normal operation process steps for the MCO.

Chapter 8.0 presents an overview of the process steps and the MCO configuration for each step. The following paragraphs discuss the pertinent thermal parameters associated with each process step as they relate to the safety basis of the system.

4.1.3.1 Water-Flooded Transportation. The water-flooded transportation phase begins at the K Basins when the SNF, loaded in the appropriate basket, is placed into the MCO. Before the SNF assemblies are placed in the fuel baskets, they are mechanically cleaned within their existing storage canisters to remove as much of the accumulated fuel corrosion product as possible. Section 4.1.2 defines the assumed condition of the fuel and scrap before they are loaded in the MCO.

Once loaded, the fuel baskets are placed into an MCO that itself has been positioned within a transportation cask. Because the entire MCO loading operation takes place under the surface of the K Basins pools, the process is considered an isothermal event. Given the size of the pools and the fact that the pools are maintained at a nominal temperature of 10 °C (50 °F), the temperature of the transportation cask, the MCO, and the SNF assemblies contained within the MCO can be assumed to be a uniform 10 °C (50 °F) when the cask-MCO is lifted out of the loading pit.

Although the MCO shield plug is installed on the MCO, it is not sealed. Pool water fills the cavity within the MCO to a height approximately 4 in. below the bottom surface of the MCO shield plug. The height of this gap is positively controlled by the design of the shield plug, which provides a "diving bell" type of enclosure around the high-efficiency particulate air (HEPA) filters. The void volume is filled with air when the shield plug is lowered into the pool. This operation not only prevents water from entering the HEPA filters, but automatically results in the creation of the desired head space within the MCO when the shield plug is inserted.

The void volume within the shield plug is verified through a combination of analytical calculations and bench testing. The design drawings and a spreadsheet-based calculation are used to compute the nominal void volume within the shield plug passages and the enclosure created above the guard plate. The spreadsheet is also used to estimate the void volume loss as a result of the approximately 8 ft of water head that will exist at the point where the shield plug is mated with the MCO. Because of its geometric complexity, the effective void volume of the HEPA filters is determined based on a physical measurement of the filter's dimensions and weight.

The feasibility of the "diving bell" approach to create the necessary void space is assured based on two facts. First, the 146-in.-long process tube, which is attached to the shield plug before the plug is inserted into the MCO, will be engaged by approximately 4 ft with the center post of the baskets before the shield plug even breaks the surface of the water. Second, the relatively tight fit between the center post and the process tube acts as a guide mechanism to ensure that the path of the shield plug remains plumb as it is lowered under water. All calculations and data supporting the effective void volume within the MCO and the cask are presented in Appendix 4B.

The annulus between the MCO and the cask is filled with deionized water before being lowered into the pool. An inflatable seal is used to prevent pool water from contaminating the water in this annulus. After removal of the inflatable seal, the final void volume in the cask-MCO annulus is created by vacuuming the water out of the gap between the MCO and the cask to the level of the shoulder in the cask wall. A lower water level is not possible because the angle and the narrow width of the gap prevents the insertion of the water removal tool.

While the cask-MCO is still in the immersion pail and the pail is in the upper position, the cask closure lid is put in position and secured. The resulting cask-MCO head space is pressurized with helium and then bled to atmospheric pressure. A cycle of pressurization and bleeding is repeated to yield an initial oxygen content within the combined void spaces that is less than the flammability limit under all circumstances in a helium-hydrogen environment. According to Bulletin 503, *Limits of Flammability of Gases and Vapors* (Coward and Jones 1952), the lower

flammability limit for oxygen concentration in a helium-hydrogen environment is approximately 4%. According to DOE Order 6430.1A, *General Design Criteria*, the design criteria for an event that is expected to occur under normal operations is 25% of the lower flammability limit, or 1% oxygen. Since the initial oxygen concentration will be reduced for every cycle, in proportion to the pressure increase imposed by the addition of the helium gas, the number of cycles required to achieve an oxygen concentration of $\leq 1\%$ must be a function of the fill pressure used. HNF-2833, *Inert Gas Requirements for Cask Loading* (Pajunen 1998b), establishes the relationship between the inerting gas pressure and the number of cycles required (see Figure 4-1). This relationship between the number of pressure-bleed cycles required versus the charge pressure used has been implemented in HNF-2169, *MCO Loading and Cask Loadout Technical Manual* (Praga 1998b).

A final pressurization step is used to establish an initial internal pressure of 5 lb/in² gauge, ± 3 lb/in². Since the MCO is vented to the cask cavity, the transportation cask is the pressure boundary under water-flooded conditions. Continued chemical reaction between the exposed uranium surfaces and the water adds hydrogen gas to the void space and pressurizes the cask cavity.

After the cask-MCO is lifted from the loading pit and the final cask closure operations are performed, it is removed from the immersion pail and placed on the transportation trailer. Removal from the pool and, to a lesser degree, from the water-filled immersion pail, will result in the temperature of the cask and its contents becoming a function of the ambient conditions, the decay and corrosion heat within the SNF assemblies, and the thermal mass of the combined system from that point forward.

The pressure conditions established at the K Basins exist until the cask-MCO arrives at the CVDF and the cask is vented. A mixture of water-flooded and helium gas-filled conditions will exist during the draining operations. Table 4-3 summarizes the pertinent thermal parameters associated with the transportation of the MCO between the K Basins, CVDF, and CSB. The 6-minute fire exposure listed in Table 4-3 is set by a risk assessment for the specific Hanford Site conditions and is the design basis for the transportation cask (Smith 2000). The 30-minute time value listed in Section 2.2.6.3 is the design basis for the MCO and is equal to the time period required by the controlling federal regulation (i.e., 10 CFR 71) should the MCO be transported offsite. All components for the MCO assembly can withstand a 30-minute fire meeting the specifications listed in Title 10, *Code of Federal Regulations*, Part 71, "Packaging and Transportation of Radioactive Material" (10 CFR 71).

4.1.3.2 Cold Vacuum Drying. The cold vacuum drying process is critical to ensuring the long-term thermal-hydraulic stability and safety basis for the MCO assembly. By removing water from the system, the source of reactants driving the chemical corrosion is reduced to a level that ensures the thermal and pressure criteria of the MCO are not violated.

Table 4-3. Thermal Aspects of Water-Flooded and Dry Transfer of K Basin Spent Nuclear Fuel.

Design parameter	Water-flooded transfer	Dry transfer
Water volume in MCO ^a	16.4 to 18.1 ft ³	0 ft ³
Gas volume in MCO ^b	0.60 ft ³	17.0 to 18.7 ft ³
Initial MCO gas backfill	Helium at 19.7 lb/in ² absolute	Helium at 25.7 lb/in ² absolute
Allowed MCO leakage rate ^c	Open to cask	10 ⁻⁵ standard cm ³ /s, air
Water volume in cask	3.88 ft ³	0 ft ³
Gas volume in cask ^d	0.48 ft ³	4.38 ft ³
Initial cask gas backfill	Helium at 19.7 lb/in ² absolute	Air at 14.7 lb/in ² absolute
Allowed cask leakage rate ^e	10 ⁻⁷ standard cm ³ /s, air	10 ⁻⁷ standard cm ³ /s, air
Nominal transfer time ^f	8 hours	14 hours
Fire accident scenario (cask exposure) ^g	6 minutes at 800 °C (1,475 °F)	6 and 30 minutes at 800 °C (1,475 °F)

^aWater volume refers to liquid, free volume. Chemically bound water and water absorbed in sludge, cracks, and crevices is not included. See Table 4-5 for derivation.

^bGas volume for water-flooded transfer assumes the void volume within shield plug with allowance for gas compression due to 8 ft of water head at point of shield plug installation.

^cLeakage rate is verified by leak test at the Cold Vacuum Drying Facility. A final leak test is conducted at the Canister Storage Building following welding of the process port covers.

^dGas volume for water-flooded transfer assumes water level in cask annulus at elevation of shoulder in cask wall (see Appendix 4B).

^eLeakage rate is verified at fabrication and annually thereafter.

^fTransfer time is defined as from time of closure at the shipping site to the time of venting at the receiving site.

^gFire accident scenario set by assessment of onsite transportation risks. See Smith, R. J., 2000, *Safety Analysis Report for Packaging, Onsite, Multi-Canister Overpack Cask*, HNF-SD-TP-SARP-017, Rev. 2, Fluor Hanford, Incorporated, Richland, Washington.

MCO = multi-canister overpack.

A combination of water-flooded, vacuum, and dry thermal conditions exists during various portions of the cold vacuum drying process as described in HNF-2356, *Spent Nuclear Fuel Project Cold Vacuum Drying Facility Operations Manual* (Irwin 1999). The MCO initially arrives at the CVDF filled with water and at a temperature that is a function of the heat load within the MCO, the ambient conditions existing during transport time from the K Basins, and the length of time between removal from the pool and arrival at the CVDF.

Following acceptance procedures, process lines are connected to the cask, and the cask-MCO headspace is vented and purged with helium. After removal of the cask lid and installation of the process hood and seal ring, tempered water (46 °C [115 °F]) is circulated

within the cask-MCO annulus as a means of preheating the MCO and its SNF contents before MCO draindown. This preheating operation lasts approximately 5.5 hours.

Draindown of the MCO is accompanied by a backfill of helium gas at 4 lb/in² gauge. While only approximately 30 minutes is needed to drain the MCO, the pre- and post-activities will expand this time frame to a total of approximately 80 minutes. Following draindown, the free water remaining within the MCO will consist of that held in puddles on horizontal surfaces, in surface films on the baskets, the MCO, and the fuel, in open cavities of broken fuel assemblies, and in cracks and crevices within damaged fuel elements and the scrap. The feasibility of removing this retained free water using the proposed vacuum drying process and equipment has been evaluated analytically and experimentally. Experimental drying tests on a prototypic MCO assembly using carbon steel cylinders as simulated fuel elements demonstrated that the free water remaining after draining can be reliably removed within the allotted time frame of the cold vacuum drying process (Ritter and McCracken 1997). The tops of the simulated fuel elements contained cavities that held water to mimic the corrosion pockets in the actual fuel elements. Additional and more realistic testing using a first-article MCO assembly and first-article process equipment skid (i.e., the vacuum purge and tempered water systems) and using simulated fuel and scrap baskets was completed in 1998 (McCracken 1999). This testing also demonstrated that the cold vacuum drying process and equipment design can successfully dry the contents of an MCO containing a nominal fuel loading in slightly more than 30 hours. An MCO containing a simulated damaged fuel loading, including simulated scrap and fines, required over 50 hours to remove the retained water loading of 22 L. An associated analytic simulation of the process accurately predicted the dryout times required to remove the retained free water.

The initial drying process step for removing this retained free water is begun using a helium purge to force out any water in the system lines. The evaporated moisture is removed from the vented gas stream using a chilled water condensing coil. Approximately 1 hour is needed to complete this step. Table 4-4 presents the design quantity for water remaining in the MCO following drain down and before initiation of the cold vacuum drying process. The base thermal-hydraulic analysis assumes that the indicated free water quantity is distributed evenly within the scrap baskets and fuel baskets, according to surface area. See Section 4.4.3.2 for further discussion. To address the sensitivity of fuel basket temperatures to early dryout, the uppermost fuel basket has its water content arbitrarily reduced to 0.1 kg. The sensitivity of the analysis to variations in the free water quantity after draining is addressed by doubling the quantity of water on the remaining fuel baskets. Beyond these results, other analyses conducted to support HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Annex B, "Cold Vacuum Drying Facility Final Safety Analysis Report" (HNF 1999), and operational procedures examined the effect of water trapped within cavities at ends of the fuel elements. These analyses showed that the time required to evaporate the free water within the MCO is extended from that shown for the base analysis. However, the system design is still capable of completing the cold vacuum drying process within the allowable time frame.

Table 4-4. Bounding Allocation of Particulate and Water in a Multi-Canister Overpack Before and After Drying at the Cold Vacuum Drying Facility.

Contributing sources of water within the MCO	Before cold vacuum drying	After cold vacuum drying	Comment
Particulate inventory (kg per basket)			
Cladding film (scrap basket)	1.54	1.54	Bounding analysis
Cladding film (fuel basket)	2.13	2.13	Bounding analysis
Oxide film (scrap basket)	0.45	0.45	Bounding analysis
Oxide film (fuel basket)	0.08	0.08	Bounding analysis
Particulate (scrap basket)	7.8	7.8	Bounding analysis
Particulate (fuel basket)	1.08	1.08	Bounding analysis
Canister particulate (fuel basket)	0.32	0.32	Bounding analysis
Generated particulate	15.00	24.8	Bounding analysis
Water inventory (kg per MCO, except where otherwise indicated)			
Free water before draining	463 to 513	N/A	See Table 4-3
Free water after draining ^a - Fuel storage basket - Scrap basket - Heel in MCO bottom	6.0/basket 1.5/basket 1.0	N/A N/A 0	
Residual free water from all sources after drying	N/A	0.200	Controlled by cold vacuum drying specification ^b
Total chemically bound water	4.56	4.64	Based on two scrap and three fuel baskets
Total water in drained/dried MCO	31.06	4.84	Based on two scrap and three fuel baskets

^aFrom Section 3.4, page 3-14, of Plys, M. G., and D. R. Duncan, 1999, *Simulation of Normal and Off-Normal Multi-Canister Overpack Behavior*, HNF-2256, Rev. 2, Fluor Daniel Hanford, Incorporated, Richland, Washington.

^bPajunen, A. L., 1998a, *Cold Vacuum Drying Residual Free Water Test Description*, HNF-1851, Rev. 2, Fluor Daniel Hanford, Incorporated, Richland, Washington.

MCO = multi-canister overpack.

N/A = not applicable.

Following the initial drying operation, a repeating cycle of vacuum drying with an accompanying 1.9 standard ft³/min helium purge followed by helium purging at a positive pressure is used to complete the drying process. The vacuum drying portion of the cycle accomplishes the majority of the moisture removal, while the helium purge portion of the cycle permits the contents within the MCO to thermally stabilize between vacuum drying cycles. The intent of the small helium purge used during the vacuum drying is to prevent hydrogen buildup from reaching potentially flammable levels should air enter the system lines through a break. The helium purge also aids in the sensible heat removal from the system.

The vacuum drying–helium purge cycle is repeated until the pressure under vacuum pump operation without helium purge indicates a sufficient level of dryness within the MCO's contents (i.e., removal of the surface water, puddles) to begin the timed process of drying the residual free water inventory contained within the cracks and crevices of the damaged fuel assemblies and scrap. This level of dryness matches the starting conditions for the single element drying tests that were conducted to verify this procedure. A timed drying process is required because a direct measurement of the residual free water inventory within an MCO after cold vacuum drying is not feasible. Instead, the cold vacuum drying process control relies on time at temperature and pressure for concluding that the residual water held in cracks and crevices within the fuel elements and scrap basket is less than 200 g. See Section 4.2.4 for further discussion.

A subsequent pressure rebound test is used to verify that no free water has been introduced into the MCO during the timed drying process (e.g., as the result of an unnoticed equipment failure). Two consecutive pressure rebound tests are required to be successfully completed before the MCO is certified as being dried.

Following vacuum drying, the MCO is cooled, pressurized with helium gas to 11 lb/in² gauge, ± 0.5 lb/in², leak checked, and prepared for shipment to the CSB. The MCO cooldown is accomplished using the facility's tempered water system to circulate 15 °C water through the cask–MCO annulus for a period of 6 hours. The leak check performed at the CVDF uses a hood over the canister to detect leakage from either the main seal or any of the process ports on the shield plug. If this test approach indicates a leak, additional testing will be required to isolate the source of the leak. A final MCO leak test is performed at the CSB following welding of the process port covers.

Although the safety analysis report for transportation (Smith 2000) does not take credit for MCO confinement, the leak rate check conducted at CVDF ensures that the MCO will provide confinement at CSB before installation of the cover cap and that oxygen will be excluded from leaking into the canister.

Table 4-4 presents the bounding allocation of chemically bound and free water remaining in an MCO after vacuum drying at the CVDF as taken from HNF-SD-SNF-TI-015 (Reilly 1998). A bounding water inventory of 4.84 kg is anticipated inside the MCO. This bounding quantity is made up of 0.2 kg of free water and 4.64 kg of chemically bound water. The bounding quantity of chemically bound water is set by the characterization work on actual N Reactor fuel and takes

no credit for the partial removal of the oxide film expected during the cleaning process in the K Basins.

4.1.3.3 Dry Transportation. The dry transportation leg begins following closure at CVDF and ends when the transportation cask is received and opened at the CSB. The MCO is sealed during this phase of transportation. All ports on the MCO shield plug are covered by blank plates to fully seal the MCO. Although the void space inside the MCO is initially filled with helium gas at a pressure of 11 lb/in² gauge, ± 0.5 lb/in² before leaving the CVDF, the actual gas environment within the MCO is a function of the heat transfer and chemical reactions that occur within the MCO during the transportation leg. Table 4-3 summarizes some of the key thermal parameters associated with the dry transportation of the MCO.

4.1.3.4 Canister Storage Building Interim Storage. Processing at the CSB begins with cask receipt and acceptance procedures at the service pit. The acceptance procedures include checking for abnormal pressure within the cask and taking appropriate recovery steps if a leaking MCO is detected. Following acceptance testing, the MCO is transferred to the MHM for in-building handling. The MHM is used to transfer the MCO to the welding station where the MCO's mechanical seal is optionally leak tested and the MCO cover cap is installed and welded in place. The MCO cover cap envelopes the mechanical closure mechanism for the MCO at the shield plug and provides a welded enclosure for the MCO and its contents.

Following weld inspections and leak testing, the MCO is transferred via the MHM to the CSB storage tubes. Each storage tube holds two MCOs in a vertical stack. Impact absorbers are placed between the MCOs and between the bottom MCO and the base of the storage tube. Approximately 5 ft of void space will exist between the top of the upper MCO and the storage tube closure plug. Provisions are included within the closure plug for monitoring the gas pressure and constituents within the tube. Air at atmospheric pressure will fill the void spaces within the storage tubes.

During the initial process checkout, six MCOs will be selected to bypass the weld station and be placed directly into the CSB storage tubes. Periodic monitoring of MCO gas pressure and constituents within the gas will be used to assess the rate and type of corrosion the spent fuel is undergoing within these selected MCOs. This monitoring is not a safety basis requirement but rather is to be used for the purpose of confirming the bounding analysis approach used for the safety basis.

4.1.4 Thermal Design Features

The MCO is designed to safely contain a variety of SNF payloads in either five or six baskets, depending on the SNF type being handled. Chapter 1.0 of this report presents a description and the design drawings of the MCO and its fuel baskets, while Chapter 2.0 provides a description of the spent fuel. The following discussion provides an overview of the important thermal design features of an MCO and its associated fuel baskets for normal operation.

The MCO shell is fabricated of stainless steel pipe with a stainless steel bottom baseplate and a machined collar at its closure end. A stainless steel shield plug, with integral HEPA filters, provides the upper closure and facilitates process line connections. A locking and lifting ring assembly is used to mechanically fasten the shield plug to the canister collar. A stainless steel cover cap, placed over the shield plug and welded to the MCO shell, is used to ensure complete confinement of the MCO contents once all processing is completed at the CSB. Support and restraint of the SNF assemblies within the MCO is accomplished using fuel baskets fabricated of stainless steel and copper and stacked in a tiered arrangement.

The overall dimensions of the MCO are 24 in. in diameter and approximately 160 in. in height. The internal MCO cavity is approximately 23 in. in diameter and 140 in. in height. The nominal empty weight of the MCO without fuel baskets is 1,553 kg. The maximum fully loaded weight for a dry MCO is 8,935 kg when loaded with Mark IV fuel elements and 8,108 kg when loaded with Mark IA fuel elements. Substitution of a scrap basket for a fuel basket will reduce the weight. Table 4-5 presents the nominal MCO weights for alternative loading configurations.

The bottom baseplate of the MCO is designed to position the lower fuel basket above the bottom of the MCO. This position promotes draining and creates a lower plenum to aid the distribution of process flows evenly across the bottom of the lower basket. The shield plug serves as the access point for various system penetrations and process connections. All penetrations through the shield plug are 1.0 in. in diameter or less and are labyrinthed to minimize radiation streaming.

A Helicoflex seal is used between the MCO shield plug and the MCO vessel wall. Metallic seals are used for the port access covers and the rupture disk. A rupture disk, installed in process port 4 on the shield plug, is used to protect the MCO from excessive pressure buildup during wet transport and cold vacuum drying processing. The rupture disk is designed for a pressure rating of 150 lb/in² gauge at 132 °C (270 °F). Analysis of the rupture disk size demonstrates that it can handle the worst-case flow rates expected during an accident scenario (Thurgood and Duncan 1997).

The SNF assemblies are placed within either a fuel storage basket or a fuel scrap basket, which is then loaded into the MCO. Copper is used in the fabrication of the scrap baskets to enhance their heat transfer capabilities. Because of height differences between the fuel assembly types, either five Mark IV baskets or six Mark IA baskets can be stacked axially within the MCO. An MCO may contain either zero, one, or two scrap baskets. The scrap basket is placed in the top location when only one scrap basket is loaded and in the top and bottom basket positions when two scrap baskets are loaded. The MCO is designed to permit internal forced flows for purging and inerting of the free volume. The MCO and its internal basket arrangement are designed to dissipate the heat from the SNF assemblies passively through a combination of conductive and radiative heat transfer with minimal reliance on internal convective flow. Figure 1-1 presents a schematic cross-sectional view of an MCO and its internals when loaded with four Mark IV fuel baskets and one scrap basket.

Table 4-5. Loaded Masses for Alternative Multi-Canister Overpack Configurations. (2 sheets)

Fuel type	Mark IV			Mark IA		
Loading description	Maximum fuel load without scrap basket	Maximum fuel load with scrap basket	Design fuel load	Maximum fuel load without scrap basket	Maximum fuel load with scrap basket	Design fuel load
Basis	270 E-length assemblies	162 E-length assemblies	Average of Mark IV assembly lengths	288 M-length assemblies	240 M-length assemblies	Average of Mark IA assembly lengths
Number of fuel baskets	5	3	4	6	4	5
Number of scrap baskets	0	2	1	0	2	1
MCO parts ^a (kg)						
- Shell	882	882	882	882	882	882
- Shield plug	700	700	700	700	700	700
- Fuel basket	518	311	415	1,287	858	1,073
- Scrap basket	0	393	196	0	568	284
Total mass of parts	2,100	2,286	2,193	2,869	3,008	2,939
Fuel materials ^b						
Assemblies ^c						
- Fuel (kg U)	6,340	3,804	4,912	4,778	3,185	3,921
- Cladding (kg Zr)	443	266	345	409	273	336
Scrap ^d						
- Fuel (kg U)	0	1,832	916	0	1,059	530
- Cladding (kg Zr)	0	128	64	0	91	45
Total masses						
- Fuel (kg U)	6,340	5,336	5,828	4,778	4,244	4,451
- Cladding (kg Zr)	443	394	409	409	364	381
Particulate ^e (kg)						
- UO ₂ •4H ₂ O	0	0	0.33	Bounded by Mark IV	Bounded by Mark IV	Bounded by Mark IV
- UO ₂	0.4	1.14	0.06			
- Al(OH) ₃	10.65	9.47	2.52			
- Adhering particulate	5.4	18.84	2.42			
- Canister particulate	1.6	0.96	0.52			
- Generated particulate						
Fuel washing to CVDF	15.0	15.0	0.08			
Cold vacuum drying	9.8	9.8	0.31			
Transport to CSB	1.5	1.5	0.23			
40-year interim storage	7.3	7.3	1.54			
Total particulate	51.65	64.01	8.01			
Water content in cladding film and particulate ^f (kg)	4.32	4.64	1.06	Bounded by Mark IV	Bounded by Mark IV	Bounded by Mark IV
Free water ^g (kg)	<0.2	<0.2	<0.2	<0.2	<0.2	<0.2
Total mass of MCO contents (kg)	6,835	5,794	6,245	5,239	4,672	4,840
Total MCO mass (kg)	8,935	8,080	8,438	8,108	7,680	7,779

Table 4-5. Loaded Masses for Alternative Multi-Canister Overpack Configurations. (2 sheets)

Fuel type	Mark IV			Mark IA		
	Maximum fuel load without scrap basket	Maximum fuel load with scrap basket	Design fuel load	Maximum fuel load without scrap basket	Maximum fuel load with scrap basket	Design fuel load
Gas void volume ^a - After drying	0.49 m ³	0.53 m ³	0.51 m ³	0.48 m ³	0.50 m ³	0.50 m ³
Scrap basket volumes, ⁱ coarse and fine scrap sections	Not applicable	0.2832 m ³ 0.0308 m ³	0.1416 m ³ 0.0154 m ³	Not applicable	0.2074 m ³ 0.0236 m ³	0.1037 m ³ 0.0118 m ³

^aDesign basis for MCO (Goldmann, L. H., 2000a, *Multi-Canister Overpack Design Report*, HNF-SD-SNF-DR-003, Rev. 3, Fluor Hanford, Incorporated, Richland, Washington.

^bMass data taken from Table 3-1 of Praga, A. N., 1998, *105-K Basin Material Design Basis Feed Description for Spent Nuclear Fuel Project Facilities, Volume 1, Fuel*, Rev. 2, Fluor Daniel Hanford, Incorporated, Richland, Washington.

^cDesign fuel loading case based on weighted average of all fuel lengths available to be placed in an MCO.

^dScrap baskets assumed loaded to the criticality mass limit for each fuel type.

^eBounding particulate values are from Slougher, J. P., 1998, *Estimates of Particulate Mass in Multi-Canister Overpacks*, HNF-1527, Rev. 2 and Reilly, M. A., 1998, *Spent Nuclear Fuel Project Technical Databook*, HNF-SD-SNF-TI-015, Rev. 6, Fluor Daniel Hanford, Incorporated, Richland, Washington. Values assume cladding film on fuel before cleaning. Mark IV particulate values bound the Mark IA values and are used here for the Mark IA cases. $\text{UO}_4 \cdot 4\text{H}_2\text{O}$ is assumed to decompose to $\text{UO}_4 \cdot 2\text{H}_2\text{O}$.

^fWater in particulate assumes no credit for removal of cladding film from surfaces of fuel assemblies.

^gValues based on Pajunen, A. L., 1998, *Cold Vacuum Drying Residual Free Water Test Description*, HNF-1851, Rev. 1, Fluor Daniel Hanford, Incorporated, Richland, Washington.

^hGas void volumes for cases without scrap baskets are based on delta weight between a fully loaded MCO filled with water and a dry MCO (see HNF-SD-SNF-DR-003 [Goldmann 2000a]). Volumes with scrap baskets are computed assuming an average basket material density of 8.04 g/cm³ and the fuel density from Table 3-1 of HNF-SD-SNF-TI-009 (Praga 1998a).

ⁱScrap basket volumes are based on drawings for Mark IV and Mark IA scrap baskets (see Figures 1-6 and 1-7).

MCO = multi-canister overpack.

A center pipe on each basket is designed to allow the use of a grapple adapter for stacking the fuel baskets within the MCO and inserting the full-length process tube that reaches the bottom of the MCO. The center pipe is designed with a center bore large enough to allow insertion of the center process tube when the upper shield plug is installed.

The process tube is used to drain the water from the MCO and to inject process gases into the MCO. When undergoing processing, gas flows from external sources, through a process connection at the top of the MCO shield plug, and into the MCO via the center process tube. The gas flow exits the process tube at the bottom of the MCO, passes radially outward through the bottom screen, and then flows up through the baseplate of the scrap or fuel baskets or is bypassed between the outer edge of the baskets and the MCO wall. An expandable metal seal around the top of the scrap basket minimizes the gas flow that bypasses the basket and maximizes the gas flow through the basket.

Some of the gas exiting the process tube can flow back up through the annular space between the center hole in the baskets and the process tube. The center pipes used for the intact fuel and scrap baskets are designed to intersect and seal with each other. This not only provides additional support, but also prevents the flow that bypasses around the locator cones from mixing with the gas flow inside the baskets at each level. The process tube, bottom screen, and locator cones have the same configurations for all basket types.

The general design of the Mark IV and Mark IA fuel baskets is shown in Figures 1-4 and 1-5. The fuel baskets are designed to support fuel assemblies vertically in individual sockets in the baseplate. An outer skirt provides additional lateral support and confinement for smaller sections of fuel assemblies. Only those portions of a combined fuel assembly (i.e., an outer and an inner element) that will fit in the basket's baseplate socket are loaded into an intact fuel basket. Solo outer or inner elements are either combined to make up a complete fuel assembly or placed in the scrap basket. Those portions of the fuel assemblies that are greater than 0.25 in. in size, but are less than approximately 3 in. in length are placed in the scrap basket. Fuel elements that are too broken up to stand upright in the intact fuel basket, or will not fit in the baseplate socket because of swelling or damage caused by corrosion also are placed in the scrap basket.

While a few pieces of SNF smaller than 0.25 in. in size may unintentionally be loaded or created as a result of loading into the basket, the design intent is to segregate pieces of this size prior to loading for handling under a separate operation. The basket heights are gauged before shield plug insertion to ensure that the inadvertent presence of small pieces of SNF is not affecting the assembly of the MCO.

The general design of the Mark IA fuel basket is shown in Figure 1-5. The basket consists of a center support tube, a 1.25-in.-thick perforated baseplate, a 2.5-in.-thick aluminum fuel rack with sockets, a 14-in.-high outer skirt, and six 3-in.-wide trapezoidal support columns. The socket diameters are large enough to allow processing gas to flow either up through the interior of the fuel elements or around the outside.

Each fuel basket is supported by six support columns that are secured to the bottom of the fuel basket. Approximately 1.5 in. of clearance exists between the top of the fuel assemblies and the lower surface of the basket above. Because of its higher initial enrichment, the Mark IA fuel is limited to 48 fuel assemblies per basket, and a large diameter center pipe is used to physically exclude the innermost circle of six fuel assemblies found in the Mark IV design.

The Mark IA scrap basket, illustrated in Figure 1-7, has a similar design, except for the absence of the socket baseplate and the addition of radial copper fins, a copper inner hexagonal chamber for holding fine particles, and a copper outer shroud. An expandable copper metal seal is affixed to the upper edge of the shroud to minimize bypass gas flow during cold vacuum drying.

The fuel basket socket design for Mark IV fuel is illustrated in Figure 1-4. Mark IV fuel assemblies have a lower initial enrichment, thus allowing 54 Mark IV fuel assemblies to be loaded per fuel basket. Because the center post on the Mark IV basket can be smaller than that required for the Mark IA basket, the six additional fuel assemblies are loaded around the center post. Each fuel basket is supported by six 1.25-in. support rods that contact the bottom baseplate of the fuel basket above it. Approximately 1.5 in. of clearance exists between the top of the fuel assemblies and the lower surface of the basket above. The other design features of the Mark IV fuel basket are similar to those discussed for the Mark IA basket.

Figure 1-6 illustrates the scrap basket for Mark IV fuel. The thermal design features of this basket are similar to those previously discussed for the Mark IA scrap basket.

Dimensions for the fuel assemblies are required as a basis for modeling the heat transfer mechanisms between the fuel assemblies and the MCO interior. Table 2-1 summarizes the geometric data for Mark IA and Mark IV fuel. The overall axial lengths of the Mark IV fuel elements range from 44.2 cm (17.4 in.) to a maximum of 66.3 cm (26.1 in.). The overall axial lengths of the Mark IA fuel elements range from 37.8 cm (14.9 in.) to a maximum of 66.3 cm (26.1 in.). Only 12 Mark IA fuel elements are 66.3 cm (26.1 in.) long, so the thermal modeling is based on a Mark IA fuel length of 53.1 cm (20.9 in.). The length of the element has a minimal effect on the thermal performance of the fuel baskets because the decay heat is proportional to volume and volume is proportional to length. Likewise, surface area (i.e., convection and radiation exchange) for any given fuel element design is also proportional to length. The maximum fuel assembly length for thermal calculations is conservative given the proportionality effect and the fact that the longer lengths increase the mean distance between the fuel and the relatively high conductive region of the basket baseplates.

4.2 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

The SNF databook, HNF-SD-SNF-TI-015 (Reilly 1998), serves as the principal source for project-approved values for key engineering, physical, and chemical parameters used in the thermal analysis and modeling of the MCO design. The information contained in the databook consists of data that are not readily available in open literature and/or are open to subjective interpretation. Values for general thermodynamic properties, such as density, specific heat,

thermal conductivity, and viscosity, not found in HNF-SD-SNF-TI-015 (Reilly 1998) are taken from standard heat transfer reference sources. The specific thermal properties used for the thermal analyses presented within this topical report are summarized in Section 4.4.1, with details provided in the underlying reference documents.

The thermal-hydraulic models used for the safety analyses utilize these thermal properties for each material that constitutes a significant heat transfer path. Properties for minor components, such as fittings and seals, are not required. The following paragraphs provide the basis used to establish the appropriate set of thermal properties for the various components of the MCO and its fuel baskets.

- The main section of the MCO shell is fabricated of SA-312 304L austenitic stainless steel. The lower end cap and the canister cover cap are SA-182 304L machined forgings. The MCO collar (the top of the MCO shell) is dual certified 304/304L stainless steel, while the locking and lifting ring is 304N stainless steel. The shield plug is fabricated of type 304L austenitic stainless steel.
- The Mark IA and IV fuel baskets are fabricated primarily of 304L stainless steel. The scrap baskets for the two fuel types also are fabricated primarily of 304L stainless steel, with the addition of ASTM B-152 C12200 copper for heat transfer enhancement.
- Miscellaneous components such as the internal screen are fabricated of 304/304L stainless steel. The rupture disk, rupture disk holder, port valves, plugs, set screws, and cover bolts are fabricated of Nitronic 60 stainless steel. The HEPA filter is fabricated of 316L stainless steel.
- The main seal between the shield plug and the canister collar uses a Helicoflex seal that is a high-strength alloy covered with a 300-series stainless steel inner jacket and a silver outer jacket. The process port covers use metal C-seals, while the process plugs are sealed with an Inconel C-seal with silver plating.
- The Mark IV and Mark IA fuel assemblies are fabricated of uranium metal with a Zircaloy-2 cladding. While the fuel assemblies are to be cleaned prior to loading in the fuel baskets, residual uranium oxides may remain adhered to the surfaces of the assemblies, lodged under defects in the cladding, or plugged in the interior gaps of the fuel assemblies.

Depending on the process step, the void spaces within the MCO may be filled with water, water vapor, helium, or a combination thereof. As a result of radiolysis and chemical reactions involving the uranium metal and uranium hydride, a variable amount of hydrogen gas may be generated and mixed with the other fluids or gases filling the void spaces. The presence of hydrogen and its concentration is a function of the operation in question, the makeup of the payload within the MCO, and the amount of moisture available to drive the reaction.

4.2.1 Scrap Basket Modeling

Since the exact geometry and makeup of the material contained within the scrap basket is not precisely known, the thermal properties for the scrap must be bounded. Two approaches are recommended for addressing heat transfer within the scrap; treat the scrap as intact fuel (i.e., as separate chunks of fuel without direct contact with other portions of the scrap) or treat it as a homogeneous, porous medium. Treating the scrap as intact fuel yields a conservative estimate of the effective heat transfer rate (unless convection from within the basket is included). Treating the scrap basket as a porous medium is more appropriate for those scrap baskets containing high reaction areas because the development of the bounding reaction area requires the presence of small pieces of fuel (Ball and Duncan 1998). Such fuel sizes will result in substantial direct contact between adjacent material within the scrap basket and between the material and the side walls of the basket.

The sensitivity of this assumption has been examined in two ways. First, a direct comparison was made of the results obtained from the HANSF¹³ code (Plys 1999a), which uses a porous bed representation for the scrap basket, and the GOTH¹⁴ code, which simulates the scrap basket payload as intact fuel assemblies with no contact. The comparison was made for the case of extended vacuum drying at the CVDF because the reduced level of conduction and convection across the void spaces seen for this condition maximizes the temperature difference between a model that assumes direct contact between the fuel scraps and one that does not. The peak scrap basket temperature obtained from the HANSF model is presented in Appendix 4A, Figure 4A-1, while the temperature obtained from the GOTH code is presented in Appendix 4A, Figure 4A-2. As seen, the difference in modeling techniques for the scrap basket results in an 8 °F (5 K) difference in the predicted peak basket temperature (i.e., 138 °F [332 K] for the HANSF code versus 146 °F [337 K] for the GOTH code). This temperature difference is within the safety margin for wet and dry transportation and for the cold vacuum drying process steps.

The second means by which a porous bed approach for simulating the scrap basket was validated was taken from the results for the proof of performance or first article testing (Crea and Fryer 1999) for the CVDF equipment. For this testing, the scrap basket payload was simulated by broken pieces of carbon steel pipe, with smaller pieces used for the fine section of the basket. Prototypic models of the MCO and baskets were used for the testing. See the report for a fuller description of the test setup. Post-test analysis of the data showed that the effective thermal conductivity obtained using an intact fuel assembly representation for the scrap basket payload had to be increased by a factor of about eight in order to achieve a match between the drying times observed in the testing and those predicted by the GOTH model. In contrast, the GOTH model was found to accurately predict the drying results for the intact fuel baskets.

¹³HANSF code developed by Fauske & Associates, Incorporated, Burr Ridge, Illinois.

¹⁴GOTH is a proprietary code to John Marvin, Inc. and is a derivative code to the GOTHIC code licensed by the Electric Power Research Institute.

Taken together, these results justify the use of the porous bed approach for simulating the thermal performance within the scrap basket while demonstrating that the maximum steady-state temperature difference between the two modeling approaches is on the order of 8 °F. Given that this level of temperature uncertainty is within acceptable levels, either modeling approach is seen as appropriate for safety basis analyses.

For a porous medium, the effective thermal conductivity (k_e) and the effective heat capacity (ρC_p) can be estimated based on the thermal properties for the solid portion and for the fluid or gas filling the void volumes, and via the porosity of the porous medium. The solid portion assumes the properties associated with uranium metal. The fluid or gas properties for water, water vapor, or helium are assumed, depending upon the operation under consideration. The effective thermal conductivity for the porous medium is calculated using Hadley's correlation, as shown by Equation 1 and documented in Table 3.1 of *Principles of Heat Transfer in Porous Media* (Kaviany 1995).

$$\frac{k_e}{k_f} = (1 - \alpha_o) \left[\frac{P f_o + (k_s/k_f)(1 - P f_o)}{1 - P(1 - f_o) + (k_s/k_f)P(1 - f_o)} \right] + \alpha_o \left[\frac{2(1 - P)(k_s/k_f)^2 + (1 + 2P)(k_s/k_f)}{(2 + P)(k_s/k_f) + 1 - P} \right] \quad (2)$$

where

k_e = effective thermal conductivity
 k_f = conductivity of the fluid phase
 α_o = parameter defined below
 P = porosity
 $f_o = 0.8 + (0.1)P$
 k_s = solid phase conductivity

and

$$\log \alpha_o = -1.084 - 6.778(P - 0.298), \quad 0.298 \leq P \leq 0.580$$

$$\log \alpha_o = 0.405 - 3.154(P - 0.0827), \quad 0.0827 \leq P \leq 0.298$$

$$\log \alpha_o = -4.898P, \quad 0 \leq P \leq 0.0827$$

This relationship provides the ratio between the effective thermal conductivity (k_e) and the conductivity of the fluid (k_f) filling the void volume within the scrap as a function of the porosity of the medium and the thermal conductivity of the solid and fluid phases (uranium fuel and water, air, helium, and other gases) present. According to Table 3.2 of Kaviany (1995), the porosity of a medium comprised of uniformly sized particles ranges from 0.26 to 0.476, depending on the method of packing.

As an alternative calculation, the porosity of the Mark IV scrap basket can be computed based on the basket containing a maximum of 980 kg of scrap. This value is based on the criticality limit for a Mark IV scrap basket (see Section 6.2.3). Based on Figure 1-6, the total available volume in the Mark IV scrap basket is approximately 157,000 cm³, with 15,400 cm³ in the fine scrap section.

The fines container has a porosity of 40% (Plys and Duncan 1998, page B-4). This porosity value is near the high end of the range recommended by Kaviani (see above). Since lower porosity values yield higher effective thermal conductivity, the higher values of porosity will yield a conservative estimate of the scrap conductivity. Given that the Mark IV fuel assembly has an average density of 0.017102 kg/cm³ (Praga 1998a), the mass contained in the fine scrap section is computed as

$$(15,400 \text{ cm}^3) (1.0 - 0.40) (0.017102 \text{ kg/cm}^3) = 158 \text{ kg.}$$

The porosity of the coarse scrap is based on a total mass of 980 kg minus the 158 kg contained in the fines section, or 822 kg. Again, this mass is assumed to be evenly distributed over the 141,600-cm³ volume of the coarse scrap section of the basket. Therefore, the porosity (P) of the coarse scrap section of the basket is computed as follows:

$$\begin{aligned} P &= 1.0 - (\text{coarse scrap mass} \div \text{density of uranium}) \\ &\quad \div (\text{volume of the coarse scrap section}) \quad (2) \\ &= 1.0 - [(980 \text{ kg} - 158 \text{ kg}) \div (0.017102 \text{ kg/cm}^3)] \\ &\quad \div [(141,600 \text{ cm}^3)] \\ &= 0.66 \end{aligned}$$

Although the actual porosity value will vary from basket to basket, the above value is an appropriate estimate for calculation purposes because the thermal-hydraulic calculations are based on full baskets. A lower scrap mass would imply both a lower reaction area and a smaller scrap volume. However, given that the effective thermal conductivity decreases with increasing porosity, rounding the porosity value up from 0.66 to 0.70 yields a value within an appropriate range and yet provides slightly higher conservatism.

The above correlation for effective conductivity versus porosity provides the best fit to the available test data (see Kaviani [1995]) for the range of parameters associated with the analyses. Uranium metal has a thermal conductivity ranging from approximately 26.8 J/m-s-K at 100 K to 39.1 J/m-s-K at 1,040 K. Minimum effective thermal conductivities are obtained with increased porosities and with low-pressure helium as the void volume gas. Maximum effective thermal conductivities are obtained with decreased porosities and with water as the void volume fluid.

The effective heat capacity of the porous entity is determined using a weighted average approach.

$$[\rho C_p]_{\text{eff}} = \rho_f C_{p_f} P + (1 - P)(\rho_s) C_{p_s} \quad (3)$$

where

ρ_f = density of the fluid
 C_{p_f} = specific heat of the fluid
 P = porosity of the porous medium
 ρ_s = density of the solid
 C_{p_s} = specific heat of the solid.

4.2.2 Thermal Source Term

The heat dissipated from the SNF payload arises from two sources: radiolytic decay and the heat of chemical reaction. While the radiolytic decay heat can be expressed as a fixed bounding value, the heat of chemical reaction must be computed as a function of the local temperature, pressure, and environment (e.g., dry air, moist air, liquid water) within the MCO because of the exponential relationship between the chemical reaction rates and the local metal temperature. Because of this relationship and the natural thermal gradients in a fuel basket, a rigidly defined bounding value in terms of watts per MCO or watts per basket covering the full range of expected conditions would need to be excessively large. Such a value would overestimate the source term for nominal conditions and result in the thermal models predicting artificially high temperatures and gas generation rates at these conditions.

Instead of bounding the chemical reaction heat with a fixed value, the approach used is to bound the parameters used to compute the reaction heat as a function of the local environment. This includes the selection of the bounding reaction rate equations, bounding values for the extent of reaction area within a given basket type and for the entire MCO, and bounding values for the enhancement factor that accounts for the irradiated condition of the N Reactor fuel and the possible presence of uranium hydride.

The contribution from each of these sources to the overall thermal source term is addressed in the following paragraphs.

4.2.2.1 Radiolytic Decay Heat Source Term. The radiolytic decay heat source term is based on the estimated MCO inventory for the various MCO loadings. Table 4-6 repeats the SNF Project-approved data for radiolytic decay heat as presented in HNF-SD-SNF-TI-015 (Reilly 1998). The radiolytic decay heat is defined for the design basis (nominal) and safety basis MCOs in terms of watts per metric ton of uranium. Since the scrap baskets contain fewer metric tons of uranium than do the fuel baskets, the bounding decay heat is lower in MCOs containing one or two scrap baskets.

Table 4-6. Radiolytic Heat Source Term.

Payload	Decay heat per MCO	Decay heat per assembly ^b	Decay heat in scrap baskets	MTU in scrap
Design basis MCO ^a	403 W	1.493 W	--	--
Safety basis MCO with 270 Mark IV assemblies	776 W	2.874 W	0 W	N/A
Safety basis MCO with 216 Mark IV assemblies and 1 scrap basket	733 W	2.874 W	112 W	0.916
Safety basis MCO with 162 Mark IV assemblies and 2 scrap baskets	690 W	2.874 W	224 W	1.832
Safety basis MCO with 228 Mark IA assemblies	585 W	2.031 W	0 W	N/A
Safety basis MCO with 240 Mark IA assemblies and 1 scrap basket	552 W	2.031 W	65 W	0.530
Safety basis MCO with 192 Mark IA assemblies and 2 scrap baskets	520 W	2.031 W	130 W	1.060
SPR ^c fuel, 1 MCO	329 W	--	0 W	N/A

^aAssumes total of 161,000 W total heat in K East and K West Basins averaged over 400 MCOs to be nominally processed.

^bComputed from 'Total decay heat' divided by the number of fuel assemblies (i.e., 270 Mark IV and 288 Mark IA assemblies).

^cOnly one MCO of this fuel type exists.

MCO = multi-canister overpack.

MTU = metric ton of uranium.

SPR = single pass reactor.

While the minimum decay heat is expected to be about one-third of the nominal value, a value of zero watts should be assumed for the purpose of computing the minimum temperatures within the MCO. Although a zero watt heat load is artificially low, it ensures that all material used in the fabrication of the MCO assembly are compatible with the worst-case minimum temperatures, and it avoids the need to certify that each MCO, including partially loaded MCOs, contains an alternative non-zero minimum value.

4.2.2.2 Chemical Reaction Heat Source Term. The heat of chemical reaction arises when the exposed uranium surfaces of the damaged SNF fuel assemblies and scrap react with the environment within the MCO. The chemical reaction heat is based on a combination of

literature-based equations and characterization work on N Reactor fuel. The supporting documents to the SNF Databook (Reilly 1998) provide details of how the literature-based equations and the characterization work are combined to yield the chemical reaction heat source term. Given the extent of that work, it is not repeated within this report.

The chemical reaction heat source term is specified in terms of Equations 4 through 15 below and the reaction areas given in Table 4-7. Equations 4 through 7 provide the stoichiometry for the various reactions expected on a per mole basis. Equations 8 through 15 relate the number of uranium milligrams reacted per hour with the local metal temperature, pressure, and environment (e.g., dry air, moist air, liquid water) and the surface area undergoing reaction. Because of the exponential relationship between the chemical reaction rates and the local conditions, the chemical reaction heat must be computed as part of any thermal simulation in order to ensure an accurate prediction of the chemical reaction heat source.

Table 4-7. Chemical Reaction Surface Area.

Payload	Scrap baskets		Storage baskets		Reaction enhancement factor	Composite reaction enhancement factor
	No.	Total geometric surface area	No.	Total geometric surface area		
Individual basket basis ^a						
Safety basis	--	45,000 cm ²	--	7,900 cm ²	10	22
Design basis	--	17,000 cm ²	--	425 cm ²	3	15
MCO basis ^b						
Safety basis	0	--	5	0 cm ² ^c	10	22
Safety basis	1	47,000 cm ²	4	33,000 cm ²	10	22
Safety basis	2	95,000 cm ²	3	25,000 cm ²	10	22
Design basis	1	17,000 cm ²	4	2,000 cm ²	3	15

^aThe minimum surface area per basket.

^bThe surface area is increased over the per basket basis in order to match the rounded MCO surface areas presented in Reilly, M. A., 1998, *Spent Nuclear Fuel Project Technical Databook*, HNF-SD-SNF-TI-015, Rev. 6, Fluor Daniel Hanford, Incorporated, Richland, Washington.

^cA conservative value of 0 cm² is used for interim storage calculations.

The reaction of uranium in air or in a water or water vapor environment is represented by

and

where the variable ' χ ' is < 0.25 . A value of ' χ ' greater than zero simply means that there may be an occasional extra oxygen atom in the matrix. However, the basic reaction remains as stated and the appropriate enthalpy characterizing the reaction is as stated.

The rate at which the indicated uranium-water or uranium-oxygen reactions occur is a function of the temperature of the uranium metal, the partial pressure of the water vapor (if present), and the surface area involved. The relationships for the chemical reaction rates are summarized in HNF-SD-SNF-TI-015 (Reilly 1998), together with supporting documentation. The relationships are expressed as Arrhenius rate law-type equations developed for the reaction rate of unirradiated uranium in various environments and temperature ranges. A summary of the relationships is as follows:

- For dry air (< 10 - 15 vppm H_2O)

$$T < 692 \text{ K, } \text{Log } K = 7.19 - 3732.0/T \quad (6)$$

$$T > 692 \text{ K, } \text{Log } K = 28.381 - 7\text{Log}(T) - 4638.2/T \quad (7)$$

- For moist air

$$T < 373 \text{ K, } 11\text{-}75\% \text{ relative humidity (RH), } \text{Log } K = 13.8808 - 5769.6/T \quad (8)$$

$$T < 373 \text{ K, } 100\% \text{ RH, } \text{Log } K = 8.333 - 3730/T \quad (9)$$

$$373 \text{ K} < T < 463 \text{ K, } < 100\% \text{ RH, } \text{Log } K = 10.566 - 4990/T + 0.3\text{Log}(P) \quad (10)$$

$$T > 463 \text{ K, } < 100\% \text{ RH, } \text{Log } K = 6.1931 - 2963/T + 0.3\text{Log}(P) \quad (11)$$

- For oxygen-free water vapor and liquid water

$$T < 373 \text{ K, } \text{Log } K = 7.634 - 3016/T \quad (12)$$

$$373 \text{ K} < T < 523 \text{ K, } \text{Log } K = 4.33 - 2144/T + 0.5\text{Log}(P) \quad (13)$$

$$523 \text{ K} < T < 735 \text{ K, } \text{Log } K = -22.915417 + 30066.5/T - 9.119078 \times 10^6/T^2 \quad (14)$$

$$735 \text{ K} < T < 923 \text{ K, } \text{Log } K = -23.905197 + 42718.8/T - 1.787581 \times 10^7/T^2 \quad (15)$$

where K is the predicted weight gain from the reaction in milligrams of oxygen per square centimeter of surface area per hour, P is the partial pressure of the water vapor in kilopascals, and T is the temperature in degrees Kelvin. HNF-SD-SNF-TI-015 (Reilly 1998) provides a set of recommendations for transitioning from one set of equations to another. The following is a summary of the recommendations for interpolating between equations.

- Moist air to oxygen-free water vapor
 - For moist air with O₂ content <10 vppm, use the oxygen-free water vapor Equations 12 to 15
 - For moist air with O₂ content >10 vppm and <100 vppm, linearly interpolate between the moist air and oxygen-free water vapor equations at the same temperature and partial pressure of water
 - For moist air with O₂ content >100 vppm, use the moist air Equations 8 to 11.
- Moist air of varying water content
 - For T < 305 K
 - RH < 75%, use the dry air Equation 6
 - 75% < RH < 100%, linearly interpolate between Equations 6 and 9
 - for 305 K < T < 368 K
 - RH < 10%, linearly interpolate between Equations 6 and 8
 - 10% < RH < 75%, use Equation 8
 - 75% < RH < 100%, linearly interpolate between Equations 8 and 9
 - for 368 K < T < T₁, where T₁ is a function of RH (see below)
 - RH < 10%, linearly interpolate between Equations 6 and 9
 - 10% < RH < 100%, use Equation 9
 - for T₁ < T < 463 K, use the greater of Equations 6 or 10
 - for 463 K < T < 692 K, use the greater of equations 6 or 11
 - for T > 692 K, use the greater of Equations 7 or 11.

The transition temperature, T₁, is computed by solving the equation for T when RH < 10%

$$\text{Log}\{10^{(7.19-3732/T)} + P[10^{(1.6604-1630.5/T)} - 10^{(0.5174-1605.5/T)}]\} - 10.566 - 0.3\text{Log}P + 4990/T = 0.$$

and from the following equation when 10% ≤ RH ≤ 100%:

$$T_1 = 1260/[2.233 + .3\text{Log}(P)]$$

Corrosion of uranium in water can lead to the formation of hydride inclusions in the corrosion product attached to the uranium metal. The long-term contribution of hydride inclusions to the reaction rate is accounted for as part of the basic uranium reaction rates given in Equations 6 through 15 above. However, the short-term contribution of hydride inclusions can be much larger.

It is estimated in Section 5 of HNF-2256, *Simulation of Normal and Off-Normal Multi-Cartridge Overpack Behavior* (Plys and Duncan 1998), that the short-term contribution of large hydride inclusions will cause rapid reaction rates up to 300 times the base reaction rate about 4% of the time. This is equivalent to assuming 4% of all reacting surfaces are undergoing

rapid reaction at any given time. The simulation of this effect within the thermal-hydraulic models is accomplished by multiplying the standard reaction rate equations above by a factor of 12 (i.e., 300 times 4%). This effect would be distributed throughout the scrap and fuel baskets in the same manner as the surface reaction area (see below).

The reaction between uranium hydride and air or uranium hydride and water is represented by:

Given that the heat of reaction is similar to that in equations (4) and (5) above, a separate accounting for hydride reactions is not necessary. Instead, the fuel reaction enhancement factor discussed below can be increased to capture the effect of hydride inclusions as well.

4.2.2.3 Surface Area Estimate. The chemical corrosion of the uranium metal occurs at the solid surface and not within the solid volume. The amount of uranium metal with surfaces exposed to the MCO environment will vary from canister to canister depending upon the amount of damaged fuel loaded in each MCO, the type and extent of the damage on the individual fuel elements, and the presence or absence of scrap baskets. As such, a safety basis estimate is required to bound the range of exposed surface areas expected for the various MCO payload configurations.

Exposed fuel surfaces arise in intact fuel assemblies primarily because of split cladding or damaged end caps. The estimate for the amount of exposed surface area per fuel basket is based on a combination of visual observations, pool chemistry measurements, and geometric calculations. Exposed surface area estimates for the scrap baskets are based on geometric calculations using the size of the baskets and the fuel pieces.

The project-approved estimate of the exposed surface area is presented in HNF-SD-SNF-TI-015 (Reilly 1998), while HNF-SD-SNF-CN-017, *Fuel Surface Area* (Ball and Duncan 1998), presents the derivations of these values. The safety basis estimate of exposed fuel surface area is 7,900 cm² per Mark IV fuel storage basket and 45,000 cm² per Mark IV scrap basket. The design basis (nominal) surface areas are 425 cm² per fuel storage basket and 17,000 cm² per scrap basket. These bounding surface areas also bound the expected exposed fuel areas for the Mark IA baskets and for the single pass reactor (SPR) fuel as well.

HNF-SD-SNF-TI-015 (Reilly 1998) also recommends bounding values for a combination of baskets in an MCO. These bounding MCO values are slightly greater than the sum of the bounding basket values. The higher values are simply the result of round-off for conservatism. Since the thermal-hydraulic models treat each basket on an individual basis, an appropriate distribution of this 'round-off' area is to apportion it to the various baskets on a weighted basis. Table 4-7 summarizes the recommended fuel surface areas for MCO loadings with zero, one, and two scrap baskets.

Because the reaction rate relationships provided in the previous section are developed for unirradiated uranium samples, an adjustment to the literature-based reaction rates is required to account for the increase in apparent surface area caused by a combination of surface roughness and swelling due to irradiation. This adjustment to the literature-based reaction rates, termed a "reaction enhancement factor", accounts for increased surface area due to radiation exposure and for surface roughness. The safety basis reaction enhancement factor of 10 matches the apparent worst-case area multiplier noted in a series of characterization tests conducted on the N Reactor fuel. A nominal factor of about three was noted in this same characterization work, and it is the recommended value for the design basis cases.

The last column in Table 4-7 presents a composite factor to be used with the literature-based reaction rate equations to account for the combined effects of the reaction enhancement factor and the hydride inclusions. While the effects of hydride inclusions will decrease with time, the use of a constant multiplier provides an additional level of conservatism.

4.2.3 Particulate Mass and Water Content

The particulate mass and water content within the MCO are major factors contributing to the thermal-hydraulic safety analysis. As such, a number of studies and analyses have been directed toward establishing bounding and best-estimate values. Establishing these estimates considered particulate quantities loaded into an MCO, those generated after loading, and the possibility of thermal decomposition during cold vacuum drying. The key studies and analyses are HNF-1523, *K-Basins Particulate Water Content, Behavior, and Impact* (Duncan and Ball 1997) and HNF-1527 (Sloughier 1998). The SNF databook (Reilly 1998) presents the project-approved values for particulate mass and water content. Table 4-5 summarizes the project-approved values for particulate and water content by fuel type and MCO loading configuration.

A significant portion of the water content is chemically bound in the form of $\text{Al}(\text{OH})_3$, $\text{UO}_3 \cdot x \text{H}_2\text{O}$ (where x is the stoichiometry number of the hydrate), and, in lesser amounts, aluminum and iron hydrates. Since much of this water content will only be released under radiolysis, the bounding MCO pressure will not be reached in the 40-year interim storage period. Section 4.4.5 provides a summary of the pressurization within the MCO under the various load conditions analyzed.

4.2.4 Cold Vacuum Drying Residual Free Water Test

Given that the water content within the MCO is a major contributing factor to the thermal-hydraulic safety analysis, a procedure has been developed to verify that the residual free water inventory of each MCO is bounded (Pajunen 1998a). The procedure is based on a model of drying water from fuel crevices packed with particulate. The model was calibrated to the drying characteristics of N Reactor fuel based on single element drying tests performed in a hot cell. The evaluation demonstrated that it is reasonable to conclude that the free water inventory of each MCO is below 200 g so long as the drying procedure is performed on each MCO. The drying

procedure is initiated by performing a pressure rise measurement at vacuum conditions after holding the MCO in helium for 4 hours to ensure fuel temperatures are equalized. The pressure rise test criterion (<2.4 torr/hr) ensures that bulk free water removal is complete and reaction rates are limited to allow holding the MCO at vacuum for long time periods.

Once these initial conditions have been established, the MCO is held at a temperature greater than 40°C (104°F) and at a pressure below 12 torr for a specified time period. The holding time period depends on the number of scrap baskets in a MCO and currently ranges from 8 hours for no scrap baskets to 20 hours for one scrap basket, to 28 hours for two scrap baskets. A single procedure holding period of 28 hours for all MCOs may ultimately be selected for the free water test procedure to eliminate potential processing errors. This procedure is implemented in the cold vacuum drying processing steps so that no MCO is transported to the CSB until it has successfully passed this criterion. MCOs that meet this criterion can be certified as having no more than 200 g of free water.

HNF-1851, *Cold Vacuum Drying Residual Free Water Test Description* (Pajunen 1998a), documents the analytical basis for setting the duration of the timed process for drying the residual free water inventory contained within the cracks and crevices of damaged fuel assemblies and scrap. The analytical drying model is calibrated using the results from eight separate single element drying tests on N Reactor fuel. The specific fuel elements chosen were selected based on visual observations in an attempt to obtain a sampling that spanned the range of fuel conditions seen in the basins. HNF-1851 (Pajunen 1998a) contains a description of the eight fuel elements selected for testing, together with the observed drying times, water release rates as a function of time, temperature, and pressure, and the total water removed. The analytical drying model developed assumes that the drying of water held in surface defects of damaged fuel assemblies can be simulated as water constrained within porous bed-filled cracks. Under this assumption, the key parameters controlling the release of the water are the number, length, and volume of the cracks holding residual free water, the pressure drop imposed on water vapor as it transverses the vapor path length in the each crack (i.e., the permeability of the porous material filling the cracks), and the temperature and pressure of the drying environment.

Since the range of data measurements taken during the single element drying tests is insufficient (e.g., no measurement of particulate amount and permeability, or the number, size, and distribution of the cracks) to experimentally determine values for each of these key parameters, it is necessary to set values for permeability and the total volume of the cracks and then use the test data to solve for the number and length of the cracks. As such, any errors introduced into the drying model by the selection of the permeability and total crack volume will be rolled into the deduced distribution for the number and length of the cracks.

Literature-based data and physical observation of the particulate matter are used to select a value of permeability that is appropriate for the particulate filled cracks and crevices found within damaged K Basin fuel. The total volume of the cracks and crevices is estimated based on the total mass of residual water removed during the drying tests and an assumed porosity of 0.6 for the cracks and crevices. Literature indicates that a porosity of 0.6 bounds the observed void fraction for packed beds consisting of a wide variety of particle shapes, including spheres, cylinders, and

granules. A sensitivity analysis using a lower porosity value of 0.35 showed the assumed value of 0.6 yielded conservative estimates for drying time (Pajunen 1998a). The value for permeability can be based on the assumed porosity, an estimated particulate grain size, and the estimated particle shape factor.

Given total crack volume, porosity, permeability, and the fluid properties for water vapor, the experimentally observed water release rate as a function of drying time from the single element drying tests is used to determine the crack and crevice distribution (i.e., the number of cracks and crevices versus the crack length) that would be necessary to achieve a match between the model-predicted water release rate and the observed rate. Each single element drying test described in HNF-1851 (Pajunen 1998a) yielded its own crack distribution, with the results from Run 6 providing the longest estimated drying time from any of the eight fuel elements tested (including Runs 7 and 8, which involved fuel elements with more severe visual damage). Once the model has been calibrated using the single element drying test results, adjustments required to permit its use in determining the residual water content within an MCO are made that consist of the following steps:

- All fuel assemblies and scrap baskets are conservatively assumed to exhibit the bounding crack distribution observed for Run 6
- The total free water content in the cracks and crevices is computed from the bounding particulate quantity given in HNF-SD-SNF-TI-015 (Reilly 1998) assuming a porosity of 0.6
- The assumed process temperature and pressure are set to those called for in the CVDF process specification (Irwin 1999).

Based on the adjustments and the bounding particulate masses listed in Table 4-4, it has been determined that an MCO must be held at 40 °C and under vacuum conditions for 8 hours if the MCO contains no scrap baskets, 20 hours if it contains one scrap basket, and 28 hours if the MCO contains two scrap baskets. After completion of this timed process, the residual free water within the MCO will be deemed to have been reduced to less than 200 g. A subsequent pressure rebound test is used to verify that no free water has been introduced into the MCO during the timed drying process (e.g., as the result of an unnoticed equipment failure). Two consecutive pressure rebound tests are required to be successfully completed before the MCO is certified as being dried.

4.3 THERMAL SPECIFICATIONS FOR MULTI-CANISTER OVERPACK COMPONENTS

This section presents the temperature and pressure limitations of the MCO, its internal components, and the payload. System-level operating limits are addressed in Chapter 12.0. The thermal limits specified below represent the inherent capability of the individual components.

Of the materials and components used in the fabrication of the MCO assembly, those that are considered the most temperature sensitive are the Helicoflex seal used between the MCO shield plug and the MCO vessel wall, the C-seals used for the port access covers, the rupture disk, and the seal between the threaded plug of the rupture disk and the shield plug port. The Helicoflex metallic seal is composed of a close-wound Inconel helical spring surrounded by a type 300 stainless steel inner lining and a thin silver outer lining. The seal has a maximum design temperature rating of 370 °C (698 °F). The C-seals are Inconel with a silver plating and are rated at 375 °C (707 °F).

Until the MCO has been processed through cold vacuum drying, a rupture disk installed in one of the process ports on the shield plug protects the MCO from destructive failure caused by excessive pressure. After cold vacuum drying, the water content is reduced to a level sufficient to allow the MCO to be sealed without concern for potential overpressurization. The MCO rupture disk is 1-in. in working diameter and fabricated from nickel 200 stainless steel. The disk has a rupture pressure rating of 150 lb/in² gauge at 132 °C (270 °F).

The remaining materials used in the fabrication of the MCO have significantly higher thermal capabilities. The type SA-312 304L stainless steel used for the MCO shell has a melting temperature above 1,400 °C (2,550 °F) and a maximum service temperature rating of 427 °C (800 °F), in accordance with Section III, Subsection NB, of the ASME Boiler and Pressure Vessel Code (ASME 1998). However, as defined in Chapter 2.0, an effective limit of 149 °C (300 °F) is used to match the structural analysis basis. Similar thermal capabilities and operating limits are established for the type 304/304L machined forging used for the shell bottom. The shield plug, the locking and lifting ring, and other miscellaneous components of the shield plug assembly also are fabricated of type 304 or 304N stainless steel.

The Mark IV and Mark IA fuel baskets are fabricated primarily of type 304L stainless steel. Because many of the components making up the basket form part of the fuel support structure within the MCO, the maximum service temperature for the baskets is 149 °C (300 °F), in accordance with the structural basis for analysis. In addition to type 304 stainless steel, the Mark IV and Mark IA scrap baskets use ASTM B-152 C12200 copper for heat transfer enhancement and ASTM B-103 UNS 51000 copper for a flow restrictor shroud. Because the copper serves no structural function once the baskets are loaded in the MCO, its appropriate temperature limit is its melting point of 1,082 °C (1,980 °F).

Although the uranium metal in the SNF assemblies has a melting temperature of 1,090 °C (1,994 °F), the lowest eutectic point between uranium and iron occurs at 600 °C (1,112 °F). As such, this lower temperature level constitutes the effective temperature limit for uranium for all operational modes. The Zircaloy-2 cladding has a melting temperature above 1,800 °C (3,272 °F). The zirconium-beryllium end caps have a melting temperature of 980 °C (1,800 °F).

4.4 THERMAL EVALUATIONS FOR NORMAL CONDITIONS

A number of thermal-hydraulic related evaluations have been conducted that support or establish the thermal safety basis for the SNF Project, from the retrieval of the fuel at the K Basins through vacuum drying and interim storage in the CSB. Some of these evaluations address an individual process step, others examine a range of process steps, while still others establish the technical basis required in the thermal-hydraulic analyses. These evaluations include HNF-1523 (Duncan and Ball 1997); HNF-1527 (Slougher 1997); HNF-SD-SNF-TI-009 (Praga 1998a); HNF-2256 (Plys and Duncan 1998); HNF-2341, *Hanford Spent Nuclear Fuel Cold Vacuum Drying Process - GOTH-SNF Simulations of Normal Operations* (Irwin et al. 1998); and HNF-SD-TP-SARP-017, *Safety Analysis Report for Packaging, Onsite, Multi-Canister Overpack Cask* (Smith 2000).

The following sections provide extracts from these evaluations that demonstrate the system safety basis under normal operating conditions. System safety for off-normal or accident conditions is addressed by the individual facility safety analysis reports and the safety analysis report for packaging for transportation. The thermal evaluations address the range of expected environmental and process conditions for normal operations and, where necessary, establish the process controls required to ensure a safe, stable, and secure operating condition under all processing steps.

4.4.1 Thermal Models

Fundamental to resolving any thermal performance issue is the development of thermal models sufficiently robust to investigate the entire range of environmental and process conditions, as well as process control requirements that may be needed to obtain a safe, stable, and secure condition. Again, safe, stable, and secure conditions involve maintaining an energy balance wherein the energy gains to the system equal the energy losses from the system. If the energy gains exceed the losses, the system temperatures will increase until the losses increase to balance the system. If the energy gains far exceed the capability of the system to reject energy, the temperature will continue to increase. The following paragraphs describe the thermal-hydraulic models and computer programs used to perform the analyses.

Although a number of one-, two-, and three-dimensional thermal-hydraulic models have been developed and applied to the analysis of the MCO assembly during various processing steps, the safety basis analysis is based on two separate computer programs and the thermal models of the MCO assembly developed for each program. The HANSF computer program was developed specifically for this project, whereas the SINDA/FLUINT¹⁵ program is a commercially available code for solving generalized thermal problems. The use of separate codes resulted from a combination of different needs, time constraints, and the desire for a means of independent

¹⁵SINDA/FLUINT is public domain code. A commercialized version is available from Cullimore and Ring Technologies, Incorporated, Littleton, Colorado.

verification of the results obtained. A brief overview of each program and the thermal models of the MCO assembly developed for each program is provided below.

Both models are based on a lumped-parameter, finite-difference approach for simulating the thermal-hydraulic performance of MCOs containing both fuel and scrap in combination with the transportation shipping cask, the CVDF, the MHM, and the CSB storage tubes. For the purposes of simulation, an MCO configured for Mark IV fuel is assumed. Specifically, the MCO contains zero, one, or two Mark IV scrap baskets and five, four, or three baskets loaded with intact, full-length Mark IV fuel assemblies. Both thermal programs have the capability for simulating steady-state and transient thermal performance.

Sections 4.4.1.1 and 4.4.1.2 provide an overview of the thermal models developed using the HANSF and SINDA/FLUINT codes, respectively. Table 4-8 provides a comparison between the modeling parameters used in both thermal models for the safety basis calculations together with a reference basis for the selected parameter values.

The Mark IV basket designs are selected for the bounding analysis basis since the lower fuel enrichment of the Mark IV fuel permits the loading of an additional six fuel assemblies per basket. These additional assemblies are placed at the center of the basket while the remainder of the basket layout is similar to the Mark IA design. The two assembly types have similar radiolytic decay heat on a volumetric basis and, except for length, nearly identical geometries, heat transfer fundamentals dictate that the Mark IV fuel baskets will exhibit the higher temperature rise between the basket edge and the peak location on the fuel. A similar situation occurs with the scrap basket designs.

The SPR fuel also is bounded by the Mark IV fuel results for similar reasons. In addition, the total inventory of SPR fuel is only 3.38 metric ton of uranium (MTU) versus a capacity of over 6 MTU for a fully loaded MCO with Mark IV fuel.

4.4.1.1 HANSF Computer Program. The HANSF computer program was developed specifically for the SNF Project. As such, its modeling capabilities and functionality are tailored to the specific MCO design and thermal problems encountered. The contents of the MCO are modeled using seven internal control volumes, as illustrated in Figure 4-2. Separate control volumes are used to represent the individual scrap and fuel baskets. Within each control volume, the model accounts for the chemical reactions, tracks water content and species concentrations, and simulates the heat transfer mechanisms. Junctions are used to connect the MCO internal control volumes and allow forced and countercurrent flow between the scrap and fuel basket regions. Upper and lower control volumes are used to represent plenum spaces above and below the scrap and fuel baskets.

Other control volumes, not shown in the figure, are used to represent the gap between the MCO and the transportation or MHM casks, or between the MCO and the weld or service pits, and between the casks or service pits and the environment. The environment is assumed to be a large volume that remains at ambient conditions, unaffected by the inflows of mass and energy.

Table 4-8. Comparison of Parameters Used In HANSF and SINDA/FLUINT Thermal Models Used For Safety Basis Analyses. (3 sheets)

Parameter	HANSF ^a	SINDA/FLUINT ^b	Reference
Heat generation parameters: power, reaction area, reaction rate			
Radiolytic decay power	Table 4-6	Table 4-6	HNF-SD-SNF-TI-015 ^c
Fuel/scrap reaction area	Table 4-7	Table 4-7	HNF-SD-SNF-TI-015 ^c
Reaction rate equations	Chapter 4.0, Equations 12 through 15 ^d	Chapter 4.0, Equations 12 through 15 ^d	HNF-SD-SNF-TI-015 ^c
Reaction rate multiplier	10 bounding 3 nominal	10 bounding 3 nominal	HNF-SD-SNF-TI-015 ^c
Rate multiplier for hydrides	12	12 dry transport 0 wet transport	HNF-SD-SNF-TI-015 ^c
Radiation heat transfer parameters: emissivity, view factors			
Oxidized uranium emissivity	0.70	0.60	0.7 ^e
Cladding emissivity	0.70	0.60	Touloukian ^a
MCO wall emissivity	0.25	0.30	0.3 ^e
Shield plug emissivity	0.25	0.30	0.3 ^e
Cask emissivity	0.25	0.30	0.3 ^e
View factors between fuel, MCO wall, fuel baskets	8x8 matrix, hand calculation	120 surfaces per assembly, VIEW ^f program	
Conduction heat transfer parameters: density, conductivity, specific heat			
<u>Density</u>			
Fuel-cladding	17,102 kg/m ³	17,184 kg/m ³	HNF-SD-SNF-TI-009 ^a
Stainless steel	8,000 kg/m ³	8,027 kg/m ³	<i>Materials Engineering^h</i>
Copper	8,954 kg/m ³	8,941 kg/m ³	<i>Materials Engineering^h</i>
<u>Conductivity</u>			
Fuel-cladding	26.9 W/m-K	27 W/m-K at 100 °C	UNI-M-61 ⁱ
Stainless steel	16 W/m-K	16.2 W/m-K at 100 °C	ASME Code ^j
Copper	398 W/m-K	396 W/m-K at 100 °C	Rohsenow et al. ^k
<u>Specific heat</u>			
Fuel-cladding	130 J/kg-K	122 J/kg-K at 100 °C	UNI-M-61 ⁱ
Stainless steel	500 J/kg-K	498 J/kg-K at 100 °C	ASME Code ^j
Copper	384 J/kg-K	381 J/kg-K at 100 °C	Touloukian ^l
<u>Fuel and scrap baskets</u>			
Fuel basket model	Mark IV	Mark IV	Note 'm'
Scrap basket model	Porous bed	Porous bed	Appendix 4A
Fine scrap porosity	0.4	0.4	Calculated value
Coarse scrap porosity	0.685	0.66	Calculated value

Table 4-8. Comparison of Parameters Used In HANSF and SINDA/FLUINT Thermal Models Used For Safety Basis Analyses. (3 sheets)

Parameter	HANSF ^a	SINDA/FLUENT ^b	Reference
Cold vacuum drying process parameters: residual water, temperatures, vacuum pump			
Free water after draining	Table 4-4	NA	Note 'n'
Chemically bound water	Tables 4-4 and 4-5	NA	
Helium purge	1.6 standard ft ³ /min	NA	
Vacuum pump	35 standard ft ³ /min	NA	
MCO annulus water	15 gal/min at 50 °C	NA	
Environmental parameters: temperature, solar			
<u>Transport</u>			
Maximum and minimum ambient temperature	115 °F	115 °F, -27 °F	Diurnal cycle from Table 4-9
Solar	Table 4-8	Table 4-8	Note 'o'
<u>CVDF and CSB operations</u>			
Maximum and minimum ambient temperature	106 °F	NA	Fixed temperature
Solar	NA	NA	

Table 4-8. Comparison of Parameters Used In HANSF and SINDA/FLUINT Thermal Models Used For Safety Basis Analyses. (3 sheets)

Parameter	HANSF ^a	SINDA/FLUINT ^b	Reference
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^aHANSF 1.2 and HANSF 1.3 beta version used for calculations; HANSF 1.2 used for all cases except case R50WA.

^bSINDA/FLUINT Version 4.1 used for all cases of wet and dry transport.

^cReilly, M. A., 1998, *Spent Nuclear Fuel Project Technical Databook*, HNF-SD-SNF-TI-015, Rev. 6, Fluor Daniel Hanford, Incorporated, Richland, Washington.

^dSee also HNF-SD-SNF-TI-015 (Reilly 1998).

^eTouloukian, Y. S., and D. P. Dewitt, 1970, *Thermophysical Properties of Matter*, "Thermal Radiative Properties Non-metallic Solids," Volume 8, p. 526, Purdue University, West Lafayette, Indiana.

^fEmergy, A. F., 1991, *VIEW - A Radiation Viewfactor Calculation Program*, Version 5.6.9, developed under NASA Grant NAG-1-41, University of Washington, Seattle, Washington.

^gPraga, A. N., 1998a, *105-K Basin Material Design Basis Feed Description for Spent Nuclear Fuel Project Facilities, Volume 1, Fuel*, HNF-SD-SNF-TI-009, Rev. 6, Fluor Daniel Hanford, Incorporated, Richland, Washington.

^hWalsh, R. (editor), 1990, *Materials Engineering*, "Materials Selector," Penton Publishing Incorporated, Cleveland, Ohio.

ⁱWeakley, E. A., 1979, *Fuels Engineering Technical Handbook*, UNI-M-61, Fuels Engineering, Fuels Production Department, UNC Nuclear Industries, Richland, Washington.

^jASME, 1998, *Boiler and Pressure Vessel Code*, Section II, Part D, "Material Properties."

^kRohsenow, W. M., J. P. Hartnett, and E. N. Ganic, 1985, *Handbook of Heat Transfer Fundamentals*, 2nd Edition, McGraw-Hill, New York, New York, p. 3-116.

^lTouloukian, Y. S., 1970, "Specific Heat - Metallic Elements and Alloys," *Thermophysical Properties Research Center Data Series*, Volume 4, Purdue University, West Lafayette, Indiana.

^mSee Section 4.1.1 for justification for using MARK IV basket as bounding configuration. HANSF model ignores basket structure in thermal model, while SINDA/FLUINT model includes the effect of the basket structure on overall thermal capacitance and conductivity. While the fuel basket design has been modified to incorporate a 'drilled hole' base plate rather than the socket hole configuration assumed in the SINDA/FLUINT model, the thermal effect is not significant. This conclusion is based on the findings presented in JMI980305-7, *Evaluation of New Intact Fuel Basket Bottom Plate Hole Design On Flow Distribution, Drying and Heating in the MCO* (Thurgood 1999).

ⁿ1.6 standard ft³/min value used in HANSF is a conservative value compared to the 1.9 standard ft³/min specified in the *Cold Vacuum Drying Facility Operations Manual*, SNF-2356, Rev. 1 (Irwin 1999).

^oHANSF uses the same insolation value versus time for both the cask lid and sides. SINDA/FLUINT varies the value of insolation as a function of time and surface orientation. No insolation is assumed for cold weather conditions.

CSB = Canister Storage Building.

CVDF = Cold Vacuum Drying Facility.

MCO = multi-canister overpack.

NA = not applicable.

Passive heat sinks within the model include the MCO wall opposite each control volume, the cask wall, the MCO bottom, and the MCO shield plug. Three nodes are used for each MCO wall section, while the shield plug is simulated with 20 nodes.

Figure 4-3 illustrates the model nodalization used to represent the SNF assemblies in the fuel baskets. As seen, a 30° segment is modeled and symmetry conditions assumed on the side boundaries. Each fuel element within the segment is modeled separately. A radiation and convection/conduction network model is used for heat transfer between the rods in the array and the MCO wall.

The adequacy of this model nodalization was examined for the HANSF code in Appendix F of HNF-2256 (Plys and Duncan 1998). This examination investigated the axial temperature gradient in a fuel element as a result of localized surface-oxidation heating. This surface oxidation was assumed to occur at one end of the element over a length that was small (3.9 in.) in comparison with the total length of the element (22 in.). The opposite end of the element was assumed to be an adiabatic surface. Heat loss is assumed to occur laterally from the element to adjacent elements or the MCO wall.

Figure 4-4 illustrates the steady-state axial temperature difference occurring for a combination of the local temperature, the rate enhancement factor, and the steam partial pressure. As expected, the temperature gradient within the element varies according to the applied oxidation heating. Low ambient temperatures yield axial gradients of less than 2 °C over the length of the element. As ambient temperature increases along with steam partial pressure and/or enhancement factor, the resulting axial gradient within the fuel element also increases. Assuming a maximum MCO wall temperature of 50 °C (according to current operating procedures), the peak fuel gradient will be less than 4 °C. As such, the model nodalization is adequate for all but the most extreme off-normal conditions.

The scrap basket is simulated as a distributed heat sink with internal conduction and radiation calculated from a debris model. The scrap basket design is divided into six equal segments with copper ribs. Each sector is further divided into an inner container for fine scrap and an outer container for coarse scrap. Figure 4-5 illustrates the scrap basket model nodalization and the assumed heat paths.

A separate one-dimensional, porous medium heat transfer model is used for the inner and outer scrap basket regions. The inner and outer scrap regions transfer heat to the copper rib in the radial direction only. A scrap porosity of 0.4 is used for the fines portion of the basket, while a porosity of 0.685 is assumed for the coarse section of the basket. Based on the discussion presented in Section 4.2.1 and Appendix 4A, this modeling approach and resolution is appropriate for safety basis analyses for transportation, CVDF, and CSB operations.

The HANSF thermal code (Plys et al. 1998b) is used for the safety basis analysis of the MCO assembly for cold vacuum drying, handling and welding operations at the CSB, and for interim storage. Table 4-8 provides a summary of the parameter values used in the HANSF thermal model. Additional information about the model, the parameter values assumed for each

simulation, and the methods used to simulate the MCO assembly and its payloads are provided in the user's manual (Plys et al. 1998b) and in HNF-2256 (Plys and Duncan 1998).

The HANSF user's manual also contains the verification and validation work conducted on the code (Plys et al. 1998b). This work includes the evaluation of the code to compute separate effects, such as evaporation, and integral testing of combined capabilities for procedures such as cold vacuum drying and interim storage. A summary of the verification and validation conducted for the HANSF code is provided in Appendix 4C.

4.4.1.2 SINDA/FLUINT Computer Program. The SINDA/FLUINT computer program is a public domain program with several commercialized versions. The SINDA/FLUINT code was developed to solve generalized lumped-parameter, finite-difference thermal problems. However, its solution technique may be applied to any problem that can be represented in a resistor-type network. The code has been evaluated and validated for simulating the thermal response of nuclear transportation packages (Glass 1988), and it has been used for the analysis of several other transportation packages for nuclear material.

SINDA/FLUINT simulates steady-state and transient temperatures using temperature-dependent material properties and heat transfer via conduction, convection, and radiation. Although the code permits one-dimensional modeling of single and multiphase flow, this capability was not utilized for this work. Complex algorithms are programmed into the solution process to compute convective heat transfer coefficients as a function of the local geometry, gas thermal properties, and temperatures or to compute the chemical reaction heat as a function of the local temperature and pressure conditions.

A major feature of the SINDA/FLUINT code is its ability to use submodels to represent common geometry sections of components such as the cask or the fuel baskets and then to combine the individual submodels to form a complete model of the cask and MCO assembly. This approach not only simplifies the modeling but reduces the verification process by minimizing the amount of original coding required to provide a complete thermal representation of the system. Precisely how this feature is used for this analysis is explained in the following paragraphs.

Five categories of analytical thermal models are used to analyze the performance of the cask-MCO and the MCO assembly for normal and hot ambient conditions of transport:

- A model of the bottom end region of the cask-MCO and MCO assembly
- A model of the typical axial midsection of the cask-MCO
- A model of the closure lid end region of the cask-MCO and MCO shield plug
- A model of a typical intact fuel basket within the MCO assembly
- A model of a scrap basket within the MCO assembly.

Figure 4-6 illustrates the layout of the thermal submodels used to evaluate the thermal performance of the cask-MCO and MCO assembly. This layout, showing the MCO arrangement with Mark IV fuel, consists of five submodels (i.e., submodels BSKT1, BSKT2, BSKT3, BSKT4, and BSKT5), together with submodels for the lid and end regions of the cask and MCO. As seen from the figure, the END submodel encompasses the lower 9.38 in. of the cask and end plug of the MCO shell. Above the END submodel are the five submodels of the MCO fuel baskets and the associated sections of the cask wall. Each of these submodels spans the 26.85 in. length of the fuel baskets. Above the last fuel basket is a thermal submodel of the MCO shield plug, the void space below the shield plug, and a 10.69-in. section of the cask wall. The LID submodel encompasses the upper 15.79-in. section of the cask wall. The thermal nodalization within each of the individual submodels is illustrated in Figures 4-7 through 4-11.

Heat transfer through and from the typical midbody section of the cask wall is simulated using an axisymmetric representation of the cask. An axisymmetric model is used because the cask thickness and the relatively small variation in heat flux from the MCO in the circumferential direction results in a minimal variation of cask wall temperatures in the circumferential direction. Figure 4-7 illustrates the five interior thermal nodes and two surface nodes used to provide this axisymmetric representation of the cask wall at the BSKT3 submodel level. A similar representation is provided at the other levels.

The thermal modeling of the MCO assembly uses a three-dimensional approach because of the variation in heat flux and the relatively low thermal mass offered by the MCO shell and fuel baskets. The MCO shell is divided into three axial sections over the axial length of each fuel basket and into 30° segments in the circumferential (azimuthal) direction. Together, nodes 70, 72, and 74 represent a 3-in.-high, 90° wide segment of the MCO. Nodes 170, 172, and 174 represent 11-in.-high segments of the MCO opposite the perforated skirt of the fuel basket, while nodes 270, 272, and 274 represent the 12.85-in.-high section of the MCO that "sees" the sections of the fuel assemblies extending above the fuel basket skirt.

Taken together, the submodels provide an axisymmetric thermal model of the MCO cask shell and a three-dimensional thermal model of the cask lid and of the MCO assembly. Heat transfer between the cask's inner surface and the MCO is assumed to be via straight conduction (i.e., $Nu = 1$) through either a water or an air medium, as appropriate for the transportation mode under consideration. The absence of hydrogen gas is conservatively (the thermal conductivity of hydrogen is 20% to 30% greater than helium) assumed for the analysis. Graybody radiation interchange across the gap also is included for those sections with a gas backfill. Heat transfer from the outer surface of the cask is via convection and radiation to the ambient environment.

Figure 4-8 provides an enlarged view of the node layout for the cask bottom end. As with the cask body, an axisymmetric representation is used. Likewise, an axisymmetric model of the end plug of the MCO vessel is used (i.e., nodes 70 to 570). This axisymmetric representation is transitioned to the three-dimensional representation used for the BSKT1 submodel. The seal port fitting and cover plate are represented by a separate submodel that is thermally connected to the axisymmetric representation of the cask end. The presence of the transport trailer is ignored for the purposes of this analysis.

Figure 4-9 illustrates the thermal node layout for the PLUG submodel. An axisymmetric representation of the shield plug is used, while the MCO shell sections are represented with three nodes around the circumference. A single thermal node (node 59) is used to represent the mean gas temperature within the void volume below the shield plug. The cask sections of this submodel (opposite nodes 870 to 974) are not shown for simplicity, but follow the same layout as illustrated in Figure 4-7. The upper section of the MCO shield plug thermal model interfaces with the thermal model of the cask lid illustrated in Figure 4-10. Because of the non-axisymmetric influence of the cask lifting trunnions (not shown in the figures), the cask lid is modeled in three dimensions. Nine thermal nodes are used to simulate the heat lost or gained from the trunnions.

The three-dimensional representation of the lid is transitioned to the axisymmetric model of the cask side wall at the PLUG submodel segment. The cask seal port fitting and cover are modeled with a separate thermal submodel that is thermally connected to the LID thermal model. Although not shown in Figure 4-10, the 12 cask closure bolts are modeled separately using four thermal nodes to represent the exposed head, the shank section (two nodes), and the threaded portion.

The basic thermal submodel of the fuel baskets and the fuel elements represents a 90° segment of a single level. The plan view of the Mark IV fuel basket assembly shown in Figure 4-11 illustrates the placement of the thermal nodes used to model the baseplate of the Mark IV fuel basket and the lower 2.6-in. section of each fuel assembly that fits within the sockets of the fuel basket baseplate. Within the 90° segment, 11 complete fuel assemblies and 5 partial fuel assemblies are represented. Symmetry conditions are assumed at the model boundaries. The enlarged view of a representative fuel assembly within the basket illustrates the placement of the thermal nodes used to simulate each fuel assembly and the surrounding basket structure.

Thermal resolution within each full fuel assembly is provided through the use of 24 thermal nodes, 12 for each element, over four axial segments within the fuel elements. Each axial segment is represented by three nodes in the circumferential direction. The lowest axial segment (shown in Figure 4-11) is 2.6 in. high. The three other axial segments, not shown, are 11 in., 9.4 in., and 3 in. high, respectively. Heat transfer from the inner to the outer element is treated as conduction and radiation (for dry transport) across the 0.21-in. gap separating the elements. Convection from the outer surface of the outer element is assumed. While the absence of hydrogen gas is conservatively assumed, the beneficial effect due to the pressure increase is included. Based on the expected axial temperature distribution within the fuel elements (see Figure 4-4 and its relevant discussion in the previous section), this level of model nodalization is adequate for the level of reaction heat generated during wet and dry transport.

The thermal model permits full flexibility in modeling the distribution of exposed surface area anywhere over the 24 nodes used to represent each fuel assembly. Since the principal mode of damage for the fuel elements is missing end caps and split cladding, the base thermal model concentrates the exposed surface area over the upper 3 in. of the fuel assemblies.

The Mark IV scrap basket is modeled as a homogeneous, porous medium using 20 thermal nodes to provide a two-dimensional, axisymmetric representation of the heat generation and

temperatures within the scrap bed. A scrap porosity of 0.4 is used for the fines portion of the basket, while a porosity of 0.66 is assumed for the coarse section of the basket. Based on the discussion presented in Section 4.2.1 and Appendix 4A, this modeling approach and resolution is appropriate for the range of temperature and heat dissipation seen during wet and dry transportation, as well as for CVDF and CSB operations. Radiolytic decay heat and reaction area are distributed on a mass-weighted basis. An additional 34 thermal nodes provide a three-dimensional, thermal representation of the base center tube, and copper side walls of the scrap basket.

In addition to computing temperatures, the thermal model computes the transient pressure rise within the MCO and cask. The calculation includes the pressure rise effects from four sources: (1) the ideal gas expansion with changes in temperature, (2) the hydrogen gas generation from chemical reactions, (3) the expansion of the water, and (4) the hydrogen dissolved in the water volume for the wet transfer phase. The void volumes and initial backfill pressures assumed are listed in Table 4-3. The pressure rise due to hydrogen gas generation is computed using the chemical reaction and gas generation rates listed in Section 4.2, while the amount of hydrogen dissolved in the water during the wet transport phase is determined using Henry's law, the volume of the water, and the partial pressure of hydrogen gas.

Table 4-8 summarizes the parameter values assumed in the SINDA/FLUINT models, together with a reference basis for the selected parameter values. Additional information about the model, the parameter values assumed for each simulation, and the methods used to simulate the MCO assembly and its payloads are provided in *Simulation of Normal Multi-Canister Overpack Behavior During Wet and Dry Transportation* (Banken 1999).

The SINDA/FLUINT code was validated against four thermal problems with known or classical solutions (Q-Metrics 1999). The specific modeling of the MCO, fuel and scrap baskets, and transportation cask was independently checked and verified as part of the quality assurance activities for the safety analysis report for packaging (Smith 2000). A summary of the verification and validation conducted for the SINDA/FLUINT code is provided in Appendix 4C.

4.4.2 Environmental Data

WHC-SD-TP-RPT-004, *Environmental Conditions for On-Site Hazardous Materials Packages* (Fadeff 1992), is the basis for the environmental conditions to be used for evaluating normal conditions. While 10 CFR 71 provides insolation values accepted by regulatory agencies for the offsite transport of nuclear material, they are not specific to the Hanford Site. The insolation data for latitude 46°N (i.e., the Hanford Site) contained in WHC-SD-TP-RPT-004 (Fadeff 1992) should be used when evaluating onsite thermal performance. Table 4-9 presents the data as a function of the time of day and the orientation of the receiving surface.

Table 4-9. Site-Specific Insolation Data for Latitude 46°N.

Time	Insolation value (Btu/h-ft ²) for vertical surface facing								Horizontal surface facing up
	N	NE	E	SE	S	SW	W	NW	
4 a.m.	0	0	0	0	0	0	0	0	0
6 a.m.	57	192	211	105	17	17	17	17	64
8 a.m.	35	173	268	208	42	32	32	32	127
10 a.m.	42	56	177	213	126	45	42	42	281
12 noon	45	45	49	120	167	120	49	45	314
2 p.m.	42	42	42	45	126	213	177	56	281
4 p.m.	35	32	32	32	42	208	268	173	127
6 p.m.	57	17	17	17	17	105	211	192	64
8 p.m.	0	0	0	0	0	0	0	0	0

The site-specific ambient temperature used for hot day conditions is based on a diurnal temperature curve for the peak summer day at the Hanford Site. The diurnal temperature cycle is also taken from WHC-SD-TP-RPT-004 (Fadeff 1992). The cycle has a peak temperature of 46 °C (115 °F), a minimum of 23.3 °C (74 °F), and a daily average temperature of 34.2 °C (93.5 °F). In comparison, 10 CFR 71 specifies a maximum, steady-state ambient air temperature value of 38 °C (100 °F). Table 4-10 presents the recommended diurnal temperature cycle to be used in computing the ambient boundary conditions.

The historical minimum ambient temperature noted for the Hanford Site is -33 °C (-27 °F). A combination of steady-state operation at this temperature with no insolation provides the worst-case minimum ambient condition. In contrast, 10 CFR 71 specifies a minimum, steady-state ambient air temperature of -29 °C (-20 °F) for analysis with concurrent structural loads (i.e., drops) and -40 °C (-40 °F) for static load conditions.

For non-safety basis analysis that seek to use the typical maximum and minimum temperatures rather than bounding values, the American Society of Heating, Ventilation, and Refrigeration Engineers (ASHRAE) recommended median of the annual extreme temperatures for the area that includes the Hanford Site may be used. According to this source (ASHRAE 1993), the median extreme winter temperature is -17 °C (2 °F) and the median extreme summer temperature is 40 °C (103.4 °F).

Table 4-10. Maximum Diurnal Ambient Temperature Cycle at Hanford Site.

Time	Temperature (°F)
12 a.m.	82
2 a.m.	78
4 a.m.	75
6 a.m.	74
8 a.m.	85
10 a.m.	97
12 p.m.	103
2 p.m.	111
4 p.m.	115
6 p.m.	113
8 p.m.	100
10 p.m.	89
Daily average	93.5

4.4.3 Thermal Results for Normal Operations

The results of a series of thermal simulations covering normal operations for the MCO, from the K Basins to interim storage, are presented in this section. Table 4-11 presents a summary of these simulations, including a brief description of each scenario, the pertinent system parameters simulated, the case label and source for the simulation, and the section within this topical report that describes and presents these results. These thermal simulations establish the safety basis for each step in the SNF processing. The results of selected sensitivity studies also are presented in this section to demonstrate the safety margin inherent in the process steps.

4.4.3.1 Water-Flooded Operations. The thermal model developed for the SINDA/FLUINT computer program is used to demonstrate the safety basis for the water-flooded transportation of the MCO assembly from the K Basins to the CVDF. The water-filled canister provides relatively high thermal conductivity between the fuel baskets and the cask exterior. Because of the vertical orientation of the canister, internal convection will limit the temperature differences between the baskets to a few degrees. Heat transfer to the external environment is based on radiation and free convection. The diurnal cycle for insolation and ambient air temperature, as presented in Tables 4-9 and 4-10, is assumed.

Table 4-11. Simulations for Normal Multi-Canister Overpack Operations. (2 sheets)

Scenario	Pertinent simulation parameters	Case label and source	Topical report section
Transport from K Basins to CVDF	Water-filled, design basis MCO with 1 scrap basket	W1SBDB, Smith 2000 ^a	Section 4.4.3.1, Figure 4-12
Transport from K Basins to CVDF	Water-filled, safety basis MCO with 1 scrap basket	W1SBSB, Smith 2000 ^a	Section 4.4.3.1, Figure 4-13
Transport from K Basins to CVDF	Water-filled, safety basis MCO with 2 scrap baskets	W2SBSB, Smith 2000 ^a	Section 4.4.3.1, Figure 4-14
Transport from K Basins to CVDF, cold weather transfer	Water-filled, safety basis MCO with no scrap baskets, steady-state ambient temperature of -27 °F and 0 °F	W0SBSB and W0SBSB-ASHRAE, Smith 2000 ^a	Sections 4.4.3.1 and 4.4.4, Figure 4-15
Cold vacuum drying with vacuum plus helium purge	1 and 2 scrap baskets	VC1A, VC2A, HNF-2256 ^b	Section 4.4.3.2, Figures 4-16 and 4-18
Cold vacuum drying with helium purge	1 and 2 scrap baskets	VC1B, VC2B, HNF-2256 ^b	Section 4.4.3.2, Figures 4-17 and 4-19
Cold vacuum drying sensitivity case	2 scrap baskets and twice free water after draining	VC2AW, HNF-2256 ^b	Section 4.4.3.2, Figure 4-20
Cold vacuum drying sensitivity case	2 scrap baskets, twice free water, and reaction rate multiplier of 50	VC2R50W, HNF-2256 ^b	Section 4.4.3.2, Figure 4-21
Cold vacuum drying sensitivity case	1 scrap basket, twice free water, 50x reaction rate, axial damage distribution	R50Wa, HNF-2256 ^b	Section 4.4.3.2, Figure 4-22
Transport from CVDF to CSB	Dry, design basis MCO with 1 scrap basket	D1SBDB, Smith 2000 ^a	Section 4.4.3.3, Figure 4-23
Transport from CVDF to CSB	Dry, safety basis MCO with 2 scrap baskets	D2SBSB, Smith 2000 ^a	Section 4.4.3.3, Figure 4-24
Transport from CVDF to CSB, cold weather transfer	Helium-filled, safety basis MCO with 1 scrap basket, zero heat generation, steady-state ambient temperature of -27 °F	D1SBSB-coldxfr, Smith 2000 ^a	Section 4.4.4
CSB operations, MHM handling	Safety basis MCO with scrap baskets	MHMC1, HNF-2256 ^b	Section 4.4.3.4, Figure 4-27
CSB operations, weld pit	Safety basis MCO with scrap baskets	WELDC1, HNF-2256 ^b	Section 4.4.3.4, Figure 4-28
CSB operations, service pit	Safety basis MCO with scrap baskets	SPITC1, HNF-2256 ^b	Section 4.4.3.4, Figure 4-26
CSB operations, interim staging	Safety basis MCO with scrap baskets	STAGC1, HNF-2256 ^b	Section 4.4.3.5, Figure 4-29

Table 4-11. Simulations for Normal Multi-Canister Overpack Operations. (2 sheets)

Scenario	Pertinent simulation parameters	Case label and source	Topical report section
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^aSmith, R. J., 2000, *Safety Analysis Report for Packaging, Onsite, Multi-Canister Overpack Cask*, HNF-SD-TP-SARP-017, Rev. 2, Fluor Hanford, Incorporated, Richland, Washington.

^bPlys, M. G., and D. R. Duncan, 1998, *Simulation of Normal and Off-Normal Multi-Canister Overpack Behavior*, HNF-2256, Rev. 1, Fluor Daniel Hanford, Incorporated, Richland, Washington.

CSB = Canister Storage Building.

CVDF = Cold Vacuum Drying Facility.

MCO = multi-canister overpack.

MHM = multi-canister overpack handling machine.

The MCO is assumed to be filled with water to a level that is slightly above the guard plate on the shield plug. This level will automatically be established as a result of the shield plug being installed under water (see Section 4.1.3.1 for a description of this operation and Appendix 4B for a calculation of the void volume). After the cask-MCO is removed from the pool, the water level in the annulus between the MCO and the cask is established via the mechanical removal of the water to a level just above the shoulder in the cask. Henry's law is used to compute the amount of hydrogen gas dissolved in the water as a function of temperature and pressure. Gas is allowed to flow between the MCO cavity and the void space between the cask and the MCO via process port 2 in the shield plug. Any flow passing through this port must first pass through the internal HEPA filter. The rupture disk installed in process port 4 will prevent an overpressurization of the MCO in the case of a situation such as a plugged HEPA filter or the inadvertent closing of port 2.

While the transportation time between the K Basins and the CVDF is expected to be 8 hours or less (including time for cask preparation and receipt), the design shipping window is set at 24 hours for normal operations (Smith 2000). To provide visualization of the cask-MCO thermal performance past the design shipping window period, all analyses have been conducted for a 62-hour time period. The results for the design basis configuration with one scrap basket are presented in Figure 4-12. For comparison purposes, the results for the thermal evaluations under the safety basis configuration of the MCO with one or two scrap baskets are presented in Figures 4-13 and 4-14, respectively. All analyses are conducted as transients since steady-state operation is not possible due to the limited void space available to contain the hydrogen gas generated. The individual contributions of radiolytic and chemical corrosion heating to the total canister heat load are presented in the figures as a function of time. A composite reaction enhancement factor of 3 is assumed.

As seen from Figure 4-12, under the design basis case with one scrap basket, all MCO and cask component temperatures remain within their respective allowable limits for the entire 62-hour analysis period. In fact, the thermal mass of the payload, water, and transport cask is sufficient to limit the change in component temperatures to 15 °C or less during the 8-hour transportation period and for the design July day ambient conditions. The effect of the sinusoidal diurnal cycle for ambient air temperature and insolation are apparent in the figure. While the cask

pressure is predicted to rise by less than 3 lb/in² during the 8-hour transportation period and by less than 5 lb/in² after 24 hours, the rate of change in pressure with time increases because of the continued heatup of the cask and its contents. The pressure within the cask is predicted to approach 36 lb/in² gauge (51 lb/in² absolute) by the end of the 62-hour transient.

Figure 4-13 repeats the transient analysis of the water-flooded transport process for the safety basis MCO with one scrap basket. Again, all MCO and cask component temperatures remain within the allowable limits for the associated component during the entire 62-hour transient. However, the higher reacting surface area and the use of the composite reaction enhancement factor of 10 combine to accelerate the pressurization of the void spaces within the cask and canister. In fact, the pressure rise seen over the nominal 8-hour transportation period is more than that seen over a 24-hour period under the design basis MCO loading. After 24 hours the pressure is seen to have risen to 66 lb/in² gauge and is predicted to reach the 150 lb/in² gauge (165 lb/in² absolute) limit for the transportation cask after approximately 36 hours.

The sudden drop in cask pressure is illustrated in the figure after 36 hours occurs as a result of the assumed venting of the cask to prevent overpressurization. This is simulated as an instantaneous pressure decrease to 20 lb/in² absolute each time the cask bleed port is opened. The simulation then assumes that the port is closed and the pressurization process is started over. The shorter time period between cycles results from the accelerated chemical corrosion due to the higher temperatures within the fuel elements with increasing time.

The 50% increase in reaction surface area for the safety basis MCO with two scrap baskets produces a similar set of results. As seen in Figure 4-14, the available time for transportation before the cask pressure limit is exceeded drops to 32 hours. The fact that the cask, MCO, and fuel temperatures are very similar to those seen for one scrap basket and for the design basis reaction area demonstrates that the bulk of the canister heating is due to the combination of radiolytic decay heat and the heat gains from the ambient conditions. The chemical reaction heat has only a slight impact on the system temperatures.

Given that the design shipping window under normal operations is 24 hours, all three bounding MCO configurations are shown as being safely transportable within this window. If the transportation time exceeds the 24-hour window, it may be necessary to relieve the cask pressure to remain within safety limits. The recovery mode available for this off-normal transport scenario is to open the cask vent port and release the pressure to the atmosphere. The canister's internal HEPA filters will limit any particulate release resulting from this operation. Cask inerting is not required after every venting cycle since air ingress will be prevented by the tools and procedures used for this step.

By relieving the pressure, an additional 10 to 16 hours of transportation time may be achieved before another pressure relief operation is necessary. See HNF-SD-TP-SARP-017 (Smith 2000) for additional discussion. At no time is a temperature excursion expected under any of the bounding MCO configurations.

Given that the primary parameter affecting the safety of the water-flooded transfer condition is pressurization from hydrogen gas, a brief discussion of the composite reaction enhancement factor used is in order. The safety basis cases examined used a composite reaction enhancement factor of 10 versus the value of 22 from HNF-SD-SNF-TI-015 (Reilly 1998). The composite factor of 10 is intended to account for increased reactivity above literature-based rates caused by the condition of the N Reactor fuel, whereas the value of 22 includes an additional factor of 12 to account for possible uranium hydride inclusions. The additional factor of 12 was used in HNF-SD-SNF-TI-015 (Reilly 1998) to bound the results of earlier reactivity testing conducted on N Reactor fuel samples in dry air environments. While this additional factor is retained in HNF-SD-SNF-TI-015 (Reilly 1998) for reasons of conservatism, both laboratory testing on N Reactor fuel in a moist, inert environment and observed pressure rises during the shipping of single elements in liquid water showed that a composite enhancement factor of 10 conservatively bounded the data. A detailed discussion of the supporting test results is provided in HNF-4165, *Uranium Oxidation Rate Summary for the Spent Nuclear fuel Project* (Pajunen 1999). Given that testing in the specific environment seen during water-flooded transport showed a composite factor of 10 to be bounding, and given the sensitivity between the assumed composite reaction factor and the permissible shipping window duration, the use of a composite factor of 22 is deemed to be overly conservative and its potential negative impact on daily operations for this transportation phase too severe to be justified.

The final safety basis case examined for water-flooded transportation is the potential for cask-MCO freeze-up caused by operations in the Hanford Site design cold day environment. Figures 4-15a and 4-15b present the results of a pair of transient analyses simulating the cold weather transportation. Figure 4-15a presents the results for transport of an MCO under a worst-case Hanford Site cold temperature of -33°C (-27°F), per Section 4.4.2. The MCO is assumed to have a decay heat loading equal to one-half of the nominal decay heat listed in Table 4-6 (i.e., 0.747 W/assembly) and no chemical reaction heat. The analysis assumes that the cask-MCO leaves the basins at time zero and at a uniform temperature of 15°C . As seen from curve 7 in Figure 4-15a, the average cask-MCO annulus water temperature is predicted to reach freezing temperature after approximately 20 hours. Because a maximum of 19 hours is assumed by CVDF operations to have elapsed before cask receipt and door closure at the CVDF and because a portion of this 19 hours will have occurred within the building enclosing the basins before shipping, the 20 hours available under this analysis bounds the worst-case time for exposure to cold weather conditions.

Because a steady-state temperature of -33°C (-27°F) represents an extreme ambient condition that will rarely be seen, an alternative analysis has been made based on the ASHRAE-recommended design winter day temperature for the area that includes the Hanford Site. According to this source (ASHRAE 1993), the median extreme winter temperature is -17°C (2°F). For conservatism, this design temperature was reduced to -18°C (0°F) for use in the analysis. Figure 4-15b depicts the transient temperature trend for this design temperature and for an MCO exhibiting zero decay and chemical reaction heat. As seen from the figure, the minimum cask-MCO temperatures, with exception of the relatively exposed cask seals, remain above freezing for the entire 24-hour period.

Based on these analyses, a safety basis can be established for any MCO when transportation occurs during ambient conditions of -18°C (0°F) or warmer. Or, alternatively, for MCOs dissipating 200 W or more may be safely transported within the 24-hour shipping window on any day. See Section 4.4.4 for a presentation of the minimum temperatures predicted for cold weather operations.

4.4.3.2 Normal Cold Vacuum Drying Operations. The thermal model developed for the HANSF computer program is used to demonstrate the safety basis for normal cold vacuum drying operations at the CVDF. The analysis is conducted for the safety basis level of reaction area for one and two scrap baskets, the safety basis decay heat, and the nominal water content expected in the MCO after draining. A nominal level of uranium hydride inclusions are also assumed. Results of the analysis are shown in Figures 4-16 through 4-21.

These analyses assume that the surface area undergoing chemical reaction (see Table 4-7) and free water content after draining (see Table 4-4) are evenly distributed within a given scrap or fuel basket. The sensitivity to these assumptions is addressed via an increase in the amount of water assumed after draining and via a separate analysis with the reaction area concentrated at the top of the assemblies. The sensitivity to the distribution of free water is addressed in the cold vacuum drying analysis via two methods. First, the water content on the upper fuel basket is arbitrarily reduced to 0.1 kg to simulate the potential for a fast fuel basket dryout and the subsequent effects on temperatures while the remaining fuel baskets provide evaporated moisture to drive the reaction rate. Second, the analyses included evaluation of double the design basis free water content following draining. See the discussion for Figures 4-20 and 4-21 below.

Although normal cold vacuum drying processing calls for an initial vacuum drying cycle lasting 8 hours, the simulations were conducted for a time period of 24 hours to better understand the potential for runaway reactions.

Figure 4-16 presents time history plots for the various model parameters for the safety basis case with one scrap basket. As seen from the figure, dryout of the MCO occurs after about 14 hours (50,000 seconds). The upper scrap basket and its contents dry out after about 2 hours and then begin to heat up because of decay and chemical reaction heat. By the end of the simulated 24-hour vacuum drying process, the upper fuel basket and the scrap basket have reached steady-state conditions. The upper fuel basket heats up faster than the lower three do because, as stated above, its water content was set at 1/60 of that for the lower baskets. This action permits the analysis to assess the peak basket temperatures that may be attained under these conditions.

The pattern of temperatures in the remaining baskets is complicated by the interplay between local evaporative cooling and radiative and convective heating from the MCO wall inward. While all of the three lower fuel baskets are dried out after 14 hours, they are still in the process of coming up to equilibrium temperatures at the end of the simulated 24-hour period under vacuum drying. Little temperature difference is seen across the scrap basket from the effects of the copper fins and the porous bed heat transfer assumed for the scrap. In contrast, the upper fuel basket exhibits up to a 20°C difference between the center and edge of the basket.

Despite operation under vacuum conditions for about three times longer than the cold vacuum drying process calls for, all temperatures remain stable. The hydrogen concentration remains well below the 4% concentration level during the entire cycle because of the vacuum pumping and the helium purge. Expected operation for the safety basis MCO with one scrap basket under an indefinite helium purge step is illustrated in Figure 4-17. As seen, all temperatures and gas concentrations again remain within allowable limits. Without the aid of vacuum drying, the scrap dryout does not occur for 13 hours, and the fuel basket dryout increases from 14 hours to 24 hours. The hydrogen concentration in the offgas stream is nearly steady at 2%.

The same indefinite vacuum and helium purge cycle steps are evaluated for the safety basis MCO with two scrap baskets, and the results are presented in Figures 4-18 and 4-19. The basket temperatures are similar to those seen with one scrap basket. The scrap baskets dry out at approximately the same time, 3 hours, and the lower two fuel baskets in about 14 hours. The upper fuel basket (tier 4) dries out nearly instantaneously due to the low initial water (0.1 kg) assumed for it. Again, all temperatures and gas concentrations remain within allowable limits throughout the evaluated time periods.

Because the amount of water remaining within the MCO following draining is an estimated value and the amount of reactant is a primary parameter in determining the temperature response under vacuum drying, a series of sensitivity analyses was performed. Figure 4-20 presents the same indefinite vacuum cycle for the safety basis MCO with two scrap baskets as presented in Figure 4-18 except that the amount of water remaining within the MCO at the start of MCO drying is doubled.

While the results show that the MCO dryout takes an additional 10 hours (i.e., 24 hours versus 14 hours), the temperatures are all stable. The scrap basket temperatures reach approximately the same level as seen with the nominal water content, while the tier 4 fuel basket increases in temperature by about 10 °C because of the extended period of reaction with the evaporated water. The tier 2 and tier 3 fuel baskets exhibit lower temperatures over the time period because of the extended period of time under evaporative cooling. The hydrogen gas content again remains at very low levels throughout the transient.

Figure 4-21 presents the results for the case with the safety basis MCO with two scrap baskets, double the residual water after draining, and a reaction rate multiplier that is increased from 10 to 50. This case evaluates the sensitivity to the combination of water content, reaction area, and/or uranium hydride content and reaction multiplier. Although, as expected, the peak scrap and fuel temperatures are increased by 10 °C to 15 °C, the figure illustrates similar results to those seen for the base case with two scrap baskets. Taken together, these results demonstrate that a substantial safety margin exists for the normal operations vacuum drying process.

The assumption of even distribution of reaction area is addressed via a separate analysis. This analysis concentrated the reaction area at the top of the fuel assembly. Figure 4-22 presents the results for the case for a safety basis MCO with one scrap basket, double the free water after draining, and an enhancement factor of 50. The results indicate that the distribution of reaction

area within a fuel basket is relatively unimportant for normal operating temperatures (i.e., temperature differences ≤ 10 °C along the assembly) but suggest that it should be addressed for the more extreme off-normal events.

4.4.3.3 Dry Transportation. The dry transportation phase for MCO handling begins when the MCO is sealed at the CVDF and ends when the MCO is received at the CSB. The thermal model developed for the SINDA/FLUINT computer program was used to analyze the safety basis for this transportation phase. The evaluation was repeated using the HANSF thermal model as a backup calculation.

The expected transportation time, including time for cask-MCO preparation at the CVDF and receipt at the CSB, is 14 hours. With allowances for variations in operating times, the total shipping window for dry transportation under normal operations is 48 hours. Operations beyond the 48-hour time period are considered off-normal. Once the MCO is sealed, the pressure rise is a function of the heat transfer balance between the MCO and the environment, the reaction area, and the amount and release rate for the reactants remaining after cold vacuum drying processing.

The MCO thermal-hydraulic performance is evaluated using a transient analysis. The starting temperatures for the transient are established based on the cooldown process step at the CVDF wherein the MCO and its contents are cooled using a 15 °C (59 °F) water flow in the cask-MCO annulus and an approximately 0.5 standard ft³/min helium purge in the MCO. This step lasts approximately 6 hours. Closure of the MCO port valves at the conclusion of this step marks the beginning of the dry transportation shipping phase. Approximately 7 hours more are required to complete the preparation of the MCO and transportation cask for shipping. Actual transportation time to the CSB is expected to be only about an hour. The remaining time in the normal shipping time frame of 14 hours is for receipt operations at the CSB.

Like the flooded transfer condition, the system temperatures are dominated by radiolytic decay heat and ambient heating. The individual contributions of heating from radiolytic decay and chemical reactions are depicted in both Figures 4-23 and 4-24.

Figure 4-23 depicts the transient thermal response of the MCO and transportation cask for the design basis configuration of the MCO with one scrap basket. The transient evaluation is conducted for a time period of 96 hours to cover the shipping window for dry transportation and to provide a visualization of the system thermal stability for periods beyond the normal shipping window. The maximum July day ambient conditions are assumed for the transient. No temperature excursions occur during the 96-hour period, and the MCO pressure remains well below its 150 lb/in² gauge pressure capability. The temperature trends for the transport cask indicate that the system is approaching a steady-state condition after 96 hours.

Under this design basis MCO configuration, neither the 200 g of free water assumed in the MCO after cold vacuum drying nor the 0.164 kg of uranium oxide hydrate water are consumed within the 96-hour time period. This is partially because of the relatively low reaction area within the design basis MCO and partially because of the nature of the locations in which any residual free water is trapped. Only a portion of the 0.164 kg of uranium oxide hydrates will thermally

decompose during the transportation period and be available for reaction. In contrast, the chemically bound water stored as aluminum trihydroxide will not be released under the thermal conditions existing during dry transport but will require long-term radiolysis to decompose.

Figure 4-24 depicts the transient thermal response of the MCO and transportation cask for the safety basis configurations for the MCO with two scrap baskets. The higher reaction areas associated with this case lead the MCO pressure to be slightly more than doubled from that seen for the design basis case.

The partial decomposition of the assumed 1.189 kg of uranium oxide hydrates is apparent in the gas contribution plot (note the decrease in the uranium oxide hydrate water content depicted by curve 7 of the upper plot and the fact that total hydrogen generation [curve 4] exceeds the 11.11 g-moles contained in the initial 200 g of residual water assumed to remain after cold vacuum drying). The fall-off in chemical reaction heat (curve 1 in lower plot) after approximately 40 hours is due to the consumption of all the free water within the MCO and the fact that the rate of reaction is limited to the rate of uranium oxide hydrate thermal decomposition.

The limited amount of water remaining in the MCO at the end of each of the transients and the fact that the water chemically bound in the aluminum trihydroxide, the aluminum and iron hydrates, and approximately one-half the water bound in the uranium oxide hydrate will not decompose at the temperatures seen in dry transportation permit a bounding pressure in the MCO to be predicted for an indefinite dry transfer event. The bounding pressure for the safety-basis case with two scrap baskets is approximately 74 lb/in² absolute, while the bounding pressure for the design basis case is approximately 48 lb/in² absolute. These bounding pressures are based on an initial backfill pressure of 25.7 lb/in² absolute, an MCO void space of approximately 0.5 m³, the MCO temperatures noted at the end of the 96-hour transient, and a total water content of 0.795 kg and 0.282 kg (excluding the chemically bound water that will not thermally decompose) for the safety basis and design basis cases, respectively. These bounding MCO pressures, plus the nearly steady-state temperature conditions seen at the end of the 96-hour transient, permit the conclusion that steady-state operation in the transportation cask can safely be accommodated from a thermal-hydraulic point of view. However, given that the annulus between the MCO and the cask is to be filled with air and given the design leakage rate of the MCO, the time spent in the transportation cask will need to be limited to 135 days or less to avoid reaching a potential flammable mixture of gases (Smith 2000).

These temperature and pressure results are supported by a similar analysis using the HANSF computer model. Figure 4-25 presents a comparison between the HANSF analysis case 'SHIPC1', as taken from Appendix T of HNF-2256 (Plys and Duncan 1998), and the SINDA/FLUINT model. This analysis case assumes one scrap basket with 45,000 cm² of reaction area and four fuel baskets with 425 cm² of reaction area each. A composite reaction enhancement factor of 22 is used. As such, the analyzed case represents an MCO configuration with one safety-basis scrap basket and four design basis intact fuel baskets.

As seen from Figure 4-25, sheet 2, the HANSF analysis predicts a maximum fuel basket temperature of approximately 390 K, or an increase of 50 K after 75 hours ($[(500,000 \text{ minus } 230,000 \text{ seconds})/3,600 \text{ seconds per hour}]$). In comparison, the SINDA/FLUINT analysis indicates a peak temperature of 110 °C (380 K) and an increase of 45 °C over the 75-hour period following the 8-hour hold at the CVDF. Further, the HANSF code predicts a maximum canister pressure of 285 kPa or 41 lb/in² absolute and a pressure increase of 13 lb/in² ($[(285 \text{ minus } 195 \text{ kPa})/6,894.76 \text{ kPa/lb/in}^2 \text{ absolute}]$) after 75 hours versus 16 lb/in² pressure increase estimated by the SINDA/FLUINT results.

The higher canister pressure rise computed by the SINDA/FLUINT model is primarily related to slight differences in the assumed release rate for the free and chemically bound water. Further, the HANSF analysis assumes a severe ambient range of 95 °F to 115 °F, while the SINDA/FLUINT analysis assumes a typical Hanford July day cycle of 74 °F to 115 °F. The cooler night temperatures assumed by the SINDA/FLUINT model slow the heatup and, thus, the maximum component temperatures noted within the canister. Taken together, these results establish the thermal-hydraulic safety basis for the MCO under dry transportation conditions for an indefinite shipping window. However, as previously discussed, a limitation of 135 days will exist for dry transportation to prevent a potential flammable mixture of gases from occurring in the cask-MCO annulus.

4.4.3.4 Receipt and Handling at the Canister Storage Building. The evaluation of the MCO thermal-hydraulic performance at the CSB consists of a series of evaluations for the various MCO environment scenarios that occur. These include receipt of the MCO in the transportation cask and its placement in the service pit, transfer within the CSB using the MHM, welding of the canister cap in the weld pit, and storage of the MCO within the CSB vaults.

Under normal operating conditions, the MCO may remain confined in the transportation cask and in the CSB service pit for up to 8 hours after its arrival from the CVDF. The safety basis for the MCO during this process step is covered by the steady-state analysis presented for the dry transportation phase (see Section 4.4.3.3). The ambient temperature assumption (i.e., 46 °C [115 °F]) and inclusion of insolation heating in the analysis for the dry transportation phase encompass the thermal environment imposed on the cask-MCO after its arrival at the CSB. An alternative evaluation of this condition was accomplished using the HANSF computer code by extending the shipping condition evaluation to 140 hours and assuming a 1-ft-thick concrete slab and an average Hanford Site ground temperature of 15 °C (60 °F). The results, presented in Figure 4-26, are seen to be approximately 30 °C higher than those depicted at the end of the 96-hour dry shipping transient in Figure 4-24 because of a combination of continued heat up in the cask and a change in boundary conditions. Under either analysis approach, the MCO temperatures are seen as being within the thermal limits for all components, and the MCO pressure remains stable and below limits.

Following receipt procedures in the service pit, the MCO is transferred to the MHM for subsequent handling and transfer operations within the CSB. The safety basis evaluation for this process step is accomplished using the HANSF computer code. The multishell MHM cask is modeled within the HANSF code as a single shell with composite material properties. The

composite properties are based on a steel shell (gamma shield), an air gap, and a Jabroc¹⁶ wood (neutron shield) shell. The potential for free or forced convection cooling within the MHM cask walls is ignored for this safety basis analysis. The external environment within the CSB is assumed to be 41 °C (106 °F). The transient response is depicted in Figure 4-27. The large thermal time constant (about 18 days) for the MHM cask is apparent in the extended transient results in that steady-state conditions are not reached within the 7-day time period evaluated. Because of the relatively poor thermal conductivity of the MHM cask wall, the peak fuel temperatures are shown as reaching 153 °C (307 °F) after 7 days in an uncooled MHM. While neither the temperatures nor the pressures show any signs of a thermal runaway during the evaluated period, extended operations in an uncooled MHM should be avoided since excessive temperatures will result before steady-state conditions are reached.

The MHM analysis was conducted using the one scrap basket and five fuel basket configuration (i.e., 733 W of decay heat). Although a five fuel basket configuration would yield 776 W, the extra 6% of decay heat is insignificant to the overall conclusion. The higher heat load can be accounted for by reducing the allowable time by 1 day.

Operations within the weld pit are similar in nature to those within the service pit in that both environments have the soil beneath the CSB as the ultimate heat sink. The difference between the situations lies in the fact that the shipping cask is not placed within the weld pit. Instead, the MCO is surrounded by a 26-in.-diameter pipe. Figure 4-28 presents the HANSF thermal model results for this situation. The initial conditions for the MCO are taken as the conditions in the service pit after 8 hours. The 72-hour transient evaluation indicates a 30 °C (54 °F) rise in MCO wall temperatures after 72 hours, with a similar, but slightly higher, increase in the peak fuel temperatures. The corresponding increase in MCO pressure observed over the same time period is from 33.4 lb/in² to 40.7 lb/in², or a 21% increase. The normal expected time in the weld pit is 8 hours. The final MCO temperatures and pressure trends demonstrate that near steady-state conditions have been reached and that all components are within the allowable limits by the end of the 72-hour transient.

4.4.3.5 Interim Storage at the Canister Storage Building. The final process step analyzed for the CSB is the interim storage operation. This analysis is conducted in a two step fashion: the first step evaluates the short-term thermal-hydraulic performance to establish the maximum temperatures within the MCO under the worst-case CSB environment, while the second step evaluates the long-term pressurization due to radiolysis and chemical reaction to establish the maximum MCO pressure over the 40-year design period for interim storage. The short-term thermal-hydraulic analysis is conducted using the HANSF computer code (Plys et al. 1998b) for a safety basis MCO. This 7-day transient is valid for the staging period that may occur between MCO receipt at the CSB and the closure operations in the weld pit, as well as during the initial portion of the long-term interim storage period.

¹⁶Jabroc is a product of Permali Gloucester, Limited.

Based on a bounding CSB vault temperature of 62 °C (144 °F), a vault tube temperature of 92 °C (198 °F) and an MCO wall temperature of 108 °C (226 °F) are derived for the bounding case by determining the temperature differential required to transfer the bounding heat load from the MCO to the vault tube and then from the vault tube to the CSB vault temperature. Fixing the MCO wall temperature at 108 °C (226 °F) as a boundary condition permits a conservative analysis of the transient heatup of the MCO fuel baskets. The analysis results, shown in Figure 4-29, show that steady-state conditions are reached in about 3 to 4 days with a peak fuel temperature of 131 °C (269 °F), and that nearly all of the uranium oxide hydrate that can decompose at the CSB temperatures does in fact decompose. The model employed for the analysis indicates that 25% of the hydrate water remains because higher temperatures are required to complete the decomposition, leading to a mass fraction of about 3% still available for radiolysis.

An MCO pressure of about 46 lb/in² absolute is obtained over the short term. The continued slight upward trend in the MCO pressure reflects the continued release of chemical reactants under radiolysis and their subsequent chemical reaction with the exposed uranium surfaces.

Long-term pressurization of the MCO will be driven by radiolysis since the temperatures within the MCO will be too low to thermally decompose the chemically bound water. Although, given sufficient time, all of the chemically bound water inventory within the MCO will eventually decompose, the actual impact on pressure depends on the decay heat, the reactive area, the amount of aluminum hydroxide in the MCO, and the time in storage. HNF-SD-SNF-TI-040, *MCO Internal Gas Composition and Pressure during Interim Storage* (Duncan and Plys 1998), evaluated these contributing factors for the full range of expected MCO payloads and determined a bounding MCO pressure of 76.4 lb/in² absolute for a 40-year storage period. Although the short-term pressure trend will fluctuate over the time period according to the ideal gas law because of changes in the CSB environment, the long-term trend will be a steadily increasing pressure over the 40-year period.

The results of both the short-term thermal-hydraulic analysis and the long-term pressurization evaluation demonstrate that the MCO temperatures and pressure will remain within the allowable limits during the entire 40-year period for interim storage.

While the design MCO wall temperature limit of 132 °C (270 °F) is not expected to be reached during interim storage, peak fuel temperature under this design MCO wall temperature may be estimated from the temperature differential presented above for the expected CSB vault conditions. No specification analysis is required because, for steady-state conditions with radiolytic decay heat only, the temperature differential between the peak fuel temperature and the MCO wall temperature is a fixed value. As such, an MCO wall temperature of 270 °F would yield a peak fuel temperature of approximately 270 °F + (268 °F - 226 °F) = 312 °F. This peak fuel temperature is within the allowable fuel temperature limits.

4.4.4 Thermal Results for Minimum Temperature Operations

The minimum temperatures within the MCO occur for a combination of minimum ambient temperatures and minimum internal heat load. A worst-case steady-state temperature of $-33\text{ }^{\circ}\text{C}$ ($-27\text{ }^{\circ}\text{F}$) is assumed based on the minimum ambient Hanford Site temperature. In addition, no solar, and one-half of the nominal radiolytic decay heat, and no chemical reaction heat is assumed to bound the probable worst-case transportation condition. All other canister handling operations will result in higher system temperatures because of active controls on the cask-MCO environment (e.g., the pool water temperature control in the K Basins, facility temperature controls in the CVDF and CSB).

HNF-SD-TP-SARP-017 (Smith 2000) provides an evaluation of the probable minimum temperatures during wet and dry transportation based on the minimum ambient temperature of $-33\text{ }^{\circ}\text{C}$ ($-27\text{ }^{\circ}\text{F}$), minimum probable internal heat loads, and the design time for transportation. Table 4-12 presents these minimum temperatures results. In all events, the minimum temperatures are within the thermal capabilities of the associated component.

The results in Table 4-12 indicate that freezing may occur during wet transfer for an MCO containing low heat loads if left exposed for 24 hours. However, given that the normal shipping window of 24 hours does not permit the cask to be exposed to ambient conditions for this entire period (see Section 4.4.3.1) and given that the minimum ambient temperature of $-33\text{ }^{\circ}\text{C}$ ($-27\text{ }^{\circ}\text{F}$) is a rare occurrence at the Hanford Site, there is little risk for cask-MCO freeze-up under normal operations. The analyses presented in Section 4.4.3.1 show that prevention of cask-MCO freeze-up during normal operations can be ensured by limiting the wet transfer of any unprotected cask to periods when the ambient temperature is above $-18\text{ }^{\circ}\text{C}$ ($0\text{ }^{\circ}\text{F}$) or, alternatively, ensuring that the MCO payload is dissipating at least 200 W of radiolytic decay heat. Because the required safety controls for this condition need to consider normal, off-normal, and accident scenarios, the specific controls to address this condition are to be provided in a future revision of HNF-SD-TP-SARP-017 (Smith 2000).

4.4.5 Maximum Multi-Canister Overpack Internal Pressures

The maximum pressures internal to the MCO under the various thermal conditions addressed in this report are summarized in this section. As stated in the above sections, some of the conditions analyzed are transient in nature and the internal pressurization of the MCO will increase as long as chemical reactions continue. Therefore, some of the pressures presented in this section are valid only for a given time period during the transient analysis.

Table 4-12. Cask and Multi-Canister Overpack Assembly Minimum Temperatures for Normal Wet and Dry Transport Conditions.

Location and condition	Design basis minimum temperatures, °C (°F)	
	Wet transfer ^a	Dry transfer ^b
Spent nuclear fuel, maximum	5 (42)	37 (99)
Spent nuclear fuel, average	4 (39)	28 (82)
Scrap basket, average	—	13 (55)
MCO sidewall, average	3 (36)	14 (56)
MCO sidewall, minimum	1 (34)	8 (45)
MCO shield plug, average	1 (33)	-5 (23)
MCO shield plug, seals	0 (32)	-6 (22)
Cask sidewall, average	1 (34)	-14 (6)
Cask sidewall, minimum	0 (32)	-16 (4)
Seal port, lid end	-7 (20)	-24 (10)
Seal port, bottom end	-2 (29)	-17 (2)
Closure lid seals, minimum	-2 (29)	-18 (0)

^aTemperatures shown are for the end of the normal of 24 hours for wet transfer assuming a zero watt MCO fuel loading, a starting temperature for cask and contents of 15 °C (59 °F), and fixed -18 °C (0 °F) ambient temperature with no solar.

^bTemperatures at the end of 48-hour shipping window with the design basis MCO and fixed -33 °C (-27 °F) ambient temperature with no solar.

MCO = multi-canister overpack.

Table 4-13 summarizes the maximum pressures noted during the analyses of the various thermal conditions to which the MCO will be subjected. Since the MCO is vented to the transportation cask during the wet transport conditions from the K Basins to the CVDF, the applicable design pressure is the pressure within the cask. Although the maximum cask pressure expected under this condition is 703 kPa (102 lb/in² absolute), this pressurization value depends on a 24-hour transport time. Transportation times above this time period will result in higher cask pressures until the cask is vented, either at the CVDF or in the field, as part of an off-normal recovery operation. HNF-SD-TP-SARP-017 (Smith 2000) addresses this possibility.

The highest MCO pressure expected under normal operations will occur during interim storage operations at the CSB. Accounting for the effects of radiolysis, oxygen and hydrogen gettering, and thermal decomposition, the bounding pressure expected after 40 years in interim storage is 76.4 lb/in² absolute. This level of pressurization yields a 2.2 safety margin, given the 150 lb/in² gauge pressure capability of the MCO design before placement of the cover cap. The safety margin will increase to 6.1 after installation of the cover cap.

Table 4-13. Cask and Multi-Canister Overpack Maximum Pressures.

Thermal load condition	MCO pressure, kPa (lb/in ² absolute)	Transport cask pressure, kPa (lb/in ² absolute)
Wet transport conditions:		
- with no scrap baskets	N/A	400 (58)
- with 1 scrap basket	N/A	558 (81)
- with 2 scrap baskets	N/A	703 (102)
CVDF operations	151 (22.0)	101 (14.7)
Dry transport conditions	324 (47)	114 (16)
CSB operations	330 (48)	N/A
Staging	312 (45)	N/A
Interim storage	527 (76.4)	N/A

Pressures for wet and dry transport are those at the end of the normal shipping window (i.e., 24 hours for wet transport and 48 hours for dry transport).

CSB = Canister Storage Building.

CVDF = Cold Vacuum Drying Facility.

MCO = multi-canister overpack.

N/A = not applicable.

4.5 CONCLUSIONS

A series of thermal-hydraulic analyses have been conducted for the various process steps to which the MCO will be subjected. These analyses demonstrate that the MCO and its contents remain within the established thermal limits for all normal operations. These analyses are based on bounding and conservative assumptions. As such, the safety basis for the MCO is established for the purposes of this topical report.

Figure 4-1. Number of Pressure-Bleed Cycles Required to Secure a Multi-Canister Overpack Oxygen Content of 2% or Less Versus Charge Pressure.

Figure 4-2. HANSF Model Control Volumes.

MP000000.CDR 6-19-98

Figure 4-3. Pin-by-Pin Nodalization Scheme.

J8889088.CDR 9-15-95

Figure 4-4. Projected Axial Gradient Within Fuel Element.

Figure 4-5. Scrap Basket Nodalization.

← Arrows indicate expected heat flow paths.
 ○ Example heat sink indexing scheme.

MF00003.COM 6-18-98

Figure 4-6. Overview of SINDA/FLUINT Thermal Submodels.

Figure 4-7. Thermal Submodel at Typical Mid-Cask Section.

Figure 4-8. Thermal Submodel at Cask Bottom.

Figure 4-9. Thermal Model at Multi-Canister Overpack Shield Plug.

Figure 4-10. Thermal Submodel at Cask Closure Lid.

Figure 4-11. Thermal Model of Tier Fuel Basket and Fuel Elements
(Lower Section).

Figure 4-12. Design Basis Case for Water-Flooded Transfer with One Scrap Basket.

Figure 4-13. Safety Basis Case for Water-Flooded Transfer and One Scrap Basket.

Figure 4-14. Safety Basis Case for Water-Flooded Transfer and Two Scrap Baskets.

Figure 4-15a. Safety Basis for Cold Weather, Water-Flooded Transfer, with No Scrap Baskets.

Figure 4-15b. Design Basis for Cold Weather, Water-Flooded Transfer, with No Scrap Baskets.

Figure 4-16. Normal Cold Vacuum Drying with One Scrap Basket.
(3 sheets)

VC1A: CVD Vac+He FFFFS U=10 UH3=12 H2O=20.6

Figure 4-16. Normal Cold Vacuum Drying with One Scrap Basket.
(3 sheets)

VC1A: CVD Vac+He FFFFS U=10 UH3=12 H2O=20.6

Figure 4-16. Normal Cold Vacuum Drying with One Scrap Basket.
(3 sheets)

Figure 4-17. Normal Cold Vacuum Drying Operations - Helium
Purge with One Scrap Basket. (3 sheets)

VC1B: Indefinite He Purge, FFFFS U=10 UH3=12 H2O=20.6

Figure 4-17. Normal Cold Vacuum Drying Operations - Helium
Purge with One Scrap Basket. (3 sheets)

VC1B: Indefinite He Purge, FFFFS U=10 UH3=12 H2O=20.6

Figure 4-17. Normal Cold Vacuum Drying Operations - Helium
Purge with One Scrap Basket. (3 sheets)

VCIB: Indefinite He Purge, FFFPS U-10 UH3=12 H2O=20.6

Figure 4-18. Normal Vacuum Drying with Two Scrap Baskets. (3 sheets)

VC2A: CVD Vac+He, SFFFS U=10 UH3=12 H2O=16.1

Figure 4-18. Normal Vacuum Drying with Two Scrap Baskets. (3 sheets)

VC2A: CVD Vac+He, SFFFS U=10 UH3=12 H2O=16.1

Figure 4-18. Normal Cold Vacuum Drying with Two Scrap Baskets. (3 sheets)

Figure 4-19. Normal Cold Vacuum Drying Operations - Helium Purge with Two Scrap Baskets. (3 sheets)

VC2B: He Purge, SFFFS U=10 UH3=12 H2O=16.1

Figure 4-19. Normal Cold Vacuum Drying Operations - Helium
Purge with Two Scrap Baskets. (3 sheets)

VC2B: He Purge, SFFFS U=10 UH3=12 H2O=16.1

Figure 4-19. Normal Cold Vacuum Drying Operations - Helium
Purge with Two Scrap Baskets. (3 sheets)

VC2B: He Purge, SFFFS U=10 UH3=12 H2O=16.1

Figure 4-20. Off-Normal Vacuum Drying with Two Scrap Baskets and Double Free Water After Draining. (3 sheets)

VC2AW: CVD Vac+He, SFFFS U=10 UH3=12 H2O=28.1

Figure 4-20. Off-Normal Vacuum Drying with Two Scrap Baskets and Double Free Water After Draining. (3 sheets)

VC2AW: CVD Vac+He, SFFFS U=10 UH3=12 H2O=28.1

Figure 4-20. Off-Normal Cold Vacuum Drying with Two Scrap Baskets and Double Free Water After Draining. (3 sheets)

Figure 4-21. Off-Normal Vacuum Drying with Two Scrap Baskets, Double Free Water After Draining, and Rate Multiplier = 50. (3 sheets)

VC2AR50W: CVD Vac+He, SFFFS U=50 UH3=0 H2O=28.1

Figure 4-21. Off-Normal Vacuum Drying with Two Scrap Baskets, Double Free Water After Draining, and Rate Multiplier = 50. (3 sheets)

VC2AR50W: CVD Vac+He, SFFFS U=50 UH3=0 H2O=28.1

Figure 4-21. Off-Normal Vacuum Drying with Two Scrap Baskets, Double Free Water After Draining, and Rate Multiplier = 50. (3 sheets)

Figure 4-22. Off-Normal Vacuum Drying, One Scrap Basket, Double Free Water, Rate Multiplier = 50, Axial Gradient Model. (3 sheets)

R50Wa: CVD Vac+He FFFFS U=50 UH3=0 H2O=38.6 Damage top

Figure 4-22. Off-Normal Vacuum Drying, One Scrap Basket, Double Free Water, Rate Multiplier = 50, Axial Gradient Model. (3 sheets)

Figure 4-22. Off-Normal Vacuum Drying, One Scrap Basket, Double Free Water, Rate Multiplier = 50, Axial Gradient Model. (3 sheets)

Figure 4-23. Design Basis Case Dry Transfer with One Scrap Basket.

Figure 4-24. Safety Basis Dry Transfer with Two Scrap Baskets.

Figure 4-25. Comparison Between HANSF & SINDA/FLUINT Analysis for Dry Transfer with One Scrap Basket. (2 sheets)

Figure 4-25. Comparison Between HANSF & SINDA/FLUINT Analysis
for Dry Transfer with One Scrap Basket. (2 sheets)

Figure 4-26. Multi-Canister Overpack Thermal Response in Service Pit.
(3 sheets)

SPITC1: CSB SERV PIT NOMINAL PREP & SHIP PFFFS

Figure 4-26. Multi-Canister Overpack Thermal Response in Service Pit.
(3 sheets)

SPITC1: CSB SERV PIT NOMINAL PREP & SHIP FFFFS S

Figure 4-26. Multi-Canister Overpack Thermal Response in Service Pit.
(3 sheets)

SPITCI: CSB SERV PIT NOMINAL PREP & SHIP FFFFS

Figure 4-27. Multi-Canister Overpack Thermal Response in Multi-Canister Overpack Handling Machine. (3 sheets)

MEMC1: MEM (Uncooled) AFTER NOMINAL PREP.SHIP.SERVICE FFFFS

Figure 4-27. Multi-Canister Overpack Thermal Response in Multi-Canister Overpack Handling Machine. (3 sheets)

MHMC1: MHM (Uncooled) AFTER NOMINAL PREP.SHIP.SERVICE FFFFS

Figure 4-27. Multi-Canister Overpack Thermal Response in Multi-Canister Overpack Handling Machine. (3 sheets)

MMHC1: MHM (Uncooled) AFTER NOMINAL PREP,SHIP,SERVICE FFFFS

Figure 4-28. Multi-Canister Overpack Thermal Response in Weld Pit.
(3 sheets)

WELDC1: WELD PIT AFTER NOMINAL PREP,SHIP,SERVICE FFFFS

Figure 4-28. Multi-Canister Overpack Thermal Response in Weld Pit.
(3 sheets)

WELDCI: WELD PIT AFTER NOMINAL PREP,SHIP,SERVICE FFFPS

Figure 4-28. Multi-Canister Overpack Thermal Response in Weld Pit.
(3 sheets)

WELDC1: WELD PIT AFTER NOMINAL PREP,SHIP,SERVICE FFFFS

Figure 4-29. Multi-Canister Overpack Thermal Response
for Canister Storage Building Staging. (3 sheets)

STAGC1: CSB STAGING NOMINAL PREP,SHIP,SERVICE FFFFS

Figure 4-29. Multi-Canister Overpack Thermal Response for Canister Storage Building Staging. (3 sheets)

STAGC1: CSB STAGING NOMINAL PREP,SHIP,SERVICE FFFFS

Figure 4-29. Multi-Canister Overpack Thermal Response for Canister Storage Building Staging. (3 sheets)

STAGC1: CSB STAGING NOMINAL PREP,SHIP,SERVICE FFFFS

APPENDIX 4A
JUSTIFICATION OF SCRAP BASKET MODELING APPROACH

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APPENDIX 4A

JUSTIFICATION OF SCRAP BASKET MODELING APPROACH

Two modeling approaches are available for simulating the heat transfer within the scrap basket payload: treat the scrap as intact fuel (i.e., as separate chunks of fuel without direct contact with other portions of the scrap) or treat it as a homogeneous, porous medium. Treating the scrap as intact fuel yields a conservative estimate of the effective heat transfer rate (unless convection from within the basket is included). Treating the scrap basket as a porous medium is appropriate and potentially more accurate, especially for those scrap baskets containing high reaction areas because the development of the bounding reaction area requires the presence of small pieces of fuel.

The sensitivity of the modeling approach for the scrap basket is examined by comparing the predicted temperature results obtained using the HANSF code,¹ which uses a porous bed representation for the scrap basket, and the GOTH² code, which simulates the scrap basket payload as intact fuel assemblies with no contact. The comparison is made for the case of extended vacuum drying operation at the Cold Vacuum Drying Facility because the reduced level of conduction and convection across the void spaces seen for this condition maximizes the temperature difference between a model that assumes direct contact between the fuel scraps and one that does not.

The peak scrap basket temperature obtained from the HANSF model for this case is presented in Figure 4A-1 while the temperature obtained from the GOTH code is presented in Figure 4A-2. As seen from the figures, the difference in modeling techniques for the scrap basket results in a 8 °F (5 K) difference in the predicted peak basket temperature (i.e., 138 °F [332 K] for HANSF versus 146 °F [337 K] for GOTH). The relatively small temperature difference seen between the two modeling approaches effectively justifies the use of either modeling approach for safety basis analyses.

¹Plys, M. G., S. J. Lee, B. Malinovic, and M. Epstein, 1998 *Hanford Spent Nuclear Fuel Safety Analysis Model HANSF1.2: User's Manual*, FAI/98-40, Rev. 0, Fauske & Associates, Incorporated, Burr Ridge, Illinois.

²GOTH is a proprietary code to John Marvin, Inc. and is a derivative code to the GOTHIC code licensed by the Electric Power Research Institute.

Figure 4A-1. Peak Scrap Basket Temperature
Using Porous Bed Model Approach.

Figure 4A-2. Peak Scrap Basket Temperature
Using Intact Fuel Model Approach.

Vacuum Pumping with 1.8SCFM Helium Purge, Uniform Water Distribution, Bounding Conditions
Mon Dec 14 12:56:01 1998
GOTH Version 4.0 - June 1998

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APPENDIX 4B

DETERMINATION OF VOID VOLUME FOR WET TRANSPORTATION

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APPENDIX 4B

DETERMINATION OF VOID VOLUME FOR WET TRANSPORTATION

The void volume within the shield plug is verified through a combination of analytical calculations and bench testing. The design drawings and an EXCEL¹ spreadsheet-based calculation are used to compute the nominal void volume within the shield plug passages and the enclosure created above the guard plate. The spreadsheet (see Table 4B-1) uses the design dimensions of the various components making up the shield plug assembly to compute the gross volumes that either contribute to or detract from the available gas volume. The spreadsheet is also used to estimate the volume loss due to gas compression at the approximately 8 ft of water head that will exist when the shield plug is mated with the multi-canister overpack (MCO).

Because of their geometric complexity, the effective void volume of the high-efficiency particulate air (HEPA) filters is determined from laboratory measurement of the filters' dimensions and weight. The cover page and pages 32, 33, and 36 from Volume 2 of laboratory notebook number HNF-N-62, which are reproduced following Table 4B-1, present the results of the dimensional and weight measurements made on a first-article filter assembly (Figure 4B-1). Based on differential weight between a water-filled and a dry filter assembly, a void volume of 76% is computed:

$$165.276 \text{ in}^3 \text{ of water} / [(2.69/2)^2 \times \pi \times 9.625 \text{ in. long} \times 4 \text{ filters}] = 76\%$$

An alternate void volume calculation is possible based on the dry weight of the filter assembly and the computed weight of its components. Based on a dry weight of 8,041 g and a density of approximately 0.29 lb/in³, a total effective assembly volume of 61 in³ is computed. Subtracting 20.9 in³ for the support structures (i.e., 10.39 in³ - 1.17 in³ + 8.44 in³ + 3.24 in³ [from the Table 4B-1 spreadsheet]) yields an effective volume based on weight of 40.1 in³ for the filters. Dividing by design gross volume for the filters (i.e., $[2.69/2]^2 \times \pi \times 9.625 \text{ in. long} \times 4 \text{ filters}$) yields a void volume ratio of 82%.

For conservatism, the lower value of 76% is used and, for further conservatism, to account for possible fabrication tolerances, a void volume of 72% is assumed for the HEPA filters used in the shield plug assembly.

The void volume in the annulus between the MCO and the cask is calculated based on the fact that this volume is created by mechanically vacuuming the water out of the gap to the level just above the shoulder in the cask wall. Thus, the void volume in the annulus is computed based on the design dimensions of the MCO and the transportation cask. The Table 4B-1 spreadsheet computes this volume as 827.02 in³.

¹EXCEL is a trademark of Microsoft, Incorporated.

Insertion of the shield plug will occur approximately 8 ft under water. Therefore, the initial gas volume within the shield plug will be compressed as the plug is lowered under water. This volume is not regained following the return of the cask-MCO to the pool surface because the gas space is vented. The Table 4B-1 spreadsheet estimates the amount of gas volume lost to this phenomenon as 227.11 in³ or nearly 18% of the original void volume. Only the void volume internal to the MCO is affected by this phenomenon.

The total void volume to be used for wet transportation calculations is 1,873.37 in³, or 0.03 m³.

Table 4B-1. Void Volume In Shield Plug (per Drawing H-2-828040, Rev. 2)
(english units - all dimensions in inches)

File: khsahm.008 6/22/1989

Component	Material	1	2.125	10.25	4.25	1342.48	Volume above guard plate & below shield plug
Guard Plate Holes	Air	1	0	0.5	2.375	5.00	Times 3 for 3 ports, vertical leg
Port Holes	Air	3	0	0.5	4.919	11.93	Times 3 for 3 ports, horizontal leg
Process Tube Port	Air	1	0	0.5	2.225	5.24	Times 3 for 3 ports, vertical leg
Process Tube Port	Air	1	0	0.3046	2.375	0.80	Process tube port
Process Tube Port	Air	1	0	0.5	5.3195	4.16	" " horizontal leg
Process Tube Port	Air	1	0	0.3046	2.225	0.85	" " vertical leg
Guard Plate Holes	Air	1	0	2.125	1.5	21.28	
Guard Plate Holes	Air	4	0	0.5	1.5	4.71	
Stabilizer Plug	Steel	-1	0	2.125	2.5	-35.47	
Stabilizer Plug	Steel	-1	0	2.575	1.75	-36.45	
Stabilizer Plug	Air	1	0.562	1.3975	1.795	8.29	Interior void volume
Process Tube	Air	1	0.6575	1.3975	1.375	6.80	" "
Process Tube	Steel	1	0	0.2865	146.945	41.41	Interior void volume
Short Draw Tube	Steel	-1	0.505	0.825	2.975	-1.27	
Short Draw Tube	Steel	-3	0	0.85	0.45	-0.92	
Filter Support	Steel	-1	2.13	2.25	1.12	-1.85	
Filter Support	Angle Iron	-1	1.75	0.25	12.38	-5.42	
Filter Support	" "	-1	2	0.25	14.38	-7.19	effective length of angled cut piece
Filter Assembly	End Plate	-2	1.88	2.4	0.25	-2.28	
Filter Assembly	Flat Iron	-1	2.5	0.25	18.63	-10.39	
Filter Assembly	Holes	4	0	0.81	0.25	1.17	
Filter Assembly	Angle Iron	-1	2	0.25	18.88	-8.44	
Filter Assembly	" "	-1	1.25	0.25	10.37	-3.24	
Filters	Filter Media	-4	0	1.35	10	-64.13	72% void factor (based on void measurements as recorded in Handbook HNF-H-82.82, page 36)
Cast-MCO					MCO Subtotal	1275.46	
Cast-MCO	Air	1	0	12.75	0.825	318.19	Head space above canister
Cast-MCO	Air	1	18.19	12.75	1	164.40	Side space at filling flange
Cast-MCO	Air	1	0.5	12.75	1.13	288.71	Side space below filling flange
Cast-MCO	Air	1	12.855	12.75	8.787	86.82	Space above shoulder in can
Cast-MCO					Cast Subtotal	827.02	

Volume Reduction Due To Water Head @ Insertion
 Net Volume After Allowance For Head Compression

-227.11 cubic inches
 1873.37 cubic inches

Figure 4B-1. First Article Filter Assembly Test. (6 sheets)

NOTEBOOK/LOGBOOK COVERSHEET			
SECTION I RECORD COPY Not for Circulation	Notebook No. <u>HNF-N-62 2</u>		
	Date of Issue <u>3/23/98</u>		Copy <u>1</u>
	Title <u>Multi-Canister Overpack (MCO) Components Testing</u>		
Author <u>Edward S. Ruff</u>	If continued from another notebook give the notebook number _____		
This is a Controlled Notebook. The assigned custodian is responsible for this book. When the book is completed, contact your Records Management Specialist for a Retention Schedule. Complete Section II of this form and return the Notebook to Document Management, A3-95.			
Responsible Custodian	Hanford ID No.	MSIN	Date Assigned
<u>Edward S. Ruff</u>	<u>0044541</u>	<u>E6-15</u>	<u>3/24/98</u>
SECTION II Complete this section prior to returning notebook to Document Management, A3-95			
Abstract:	(Give brief description of notebook contents)		
Period Covered:	(Inclusive dates - Month/Day/Year) _____		
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Custodian's Signature and Date _____			
Retention Schedule:	Specialist Concurrence Name	Custodian's Manager's Name	

SD-8001-258 (12/97)

Figure 4B-1. First Article Filter Assembly Test. (6 sheets)

Aug. 13, 1998. 305 ETZ.

Measurements by MCO Hydraulic
Torque Wrench Drive Socket.

~~Sketch~~ Aug 13, 1998.

3/5
Thread. NC

JUNE 17, 1999. Thursday. 10:45 AM.
305 EQUIPMENT TESTING LABORATORY.

TASK: MEASUREMENT OF MCO FILTER ASSEMBLY

SOURCE: FILTER ASSEMBLY PROVIDED BY
OREGON IRON WORKS

SKETCH OF ASSEMBLY:

A = OD = 2.690"

B = LOA = 9 7/64"

F = Inlet
wall
thickness = 0.125

E = 9 7/64" Long - from
end of cartridge to outer
surface of inlet.

G = Depth to
end plug = 9 1/64"
from outer surface

C = Inlet ID = 1.211" ϕ
D = Inlet length = 1.125"

Figure 4B-1. First Article Filter Assembly Test. (6 sheets)

33

June 17, 1999. continued.

1. Zero Dial Caliper.

MITUTOYO #505-637-50. - Model No.
.001" accuracy.

Serial No. 3178034.

2. Machinists Metal Straight Rule -
inspect for damage - OK.Measured exterior dimensions of
filter cartridge & indicated on
page 32 (previous page).Edward S. Ruff
EDWARD S. RUFF
TEST ENGINEER

June 17, 1999

Jerry W. Oliver
JERRY W. OLIVER
Plant Eng.

JUNE 17, 1999

Figure 4B-1. First Article Filter Assembly Test. (6 sheets)

36

June 22, 1999. 305 ETZ.

MCO FILTER CARTRIDGE WEIGHT / VOID VOLUME.

Scale Used = Mettler PK 36

Standard Lab No = 347-06-01-004.

Cal. Expired 1-31-99.

RAW DATA:WEIGHT (GRAMS)CONFIGURATION

8041

FILTER CARTRIDGE ASSEMBLY - NO TAPE
Empty, Dry

8073

FILTER CARTRIDGE ASSY - WITH TAPE
Empty, Dry After 20 hours
in chamber at 270°F.

8448

FILTER CARTRIDGE ASSY - WITH TAPE
AFTER DRAIN (manual) and
compressed air blow-out
(No temp. drying).

10,780 *

FILTER CARTRIDGE ASSY - WITH TAPE
FULL OF WATER INSIDE FILTER
CYLINDERS. EXCESS WIPED OFF
OF OUTER SURFACE.

* Note: Filter Cartr. Assy. was immersed to
1 ft below surface, held under water,
periodically shaken until all bubbles
(i.e. air release) had ceased. Wet weights made
on 6/17/99, Dry weights recorded on 6/22/99

Edward S. Ruff

TEST ENGINEER

June 22, 1999

Jerry W. Oliver

PMT ENGINEER

JUNE 23, 1999

Figure 4B-1. First Article Filter Assembly Test. (6 sheets)

This calculation determines the estimated void volume of the MCO filter cartridge assembly.

It is based on measurements and tests performed on June 17 thru July 22, 1999 at the 305 Bldg.

File Name: c:\math\mcotest\FILTERVolume.mcd

First, a bounding calculation on internal volume of the filter cartridges is made, based on measurements of the external dimensions.

This is done as a cautionary double-check on the experimental data. That is, the internal void volume of the filter cartridges, as measured by the water dunk test, could never exceed the computed volume based on external dimensions of the filter cartridge.

Ref: HNF-N-62, Volume 2, pages 32, 33, and 36. (hardbound lab notebook, original data)

$$OD := 2.690 \text{ in}$$

$$WallThickness := 0.0625 \text{ in}$$

$$InternalLengthToEndPlug := \left[9 + \left(\frac{27}{64} \right) \right] \cdot \text{in}$$

$$InternalVolumeEachFilterCartridge_{max} := \pi \left[\frac{(OD - 2 \cdot WallThickness)^2}{4} \right] \cdot (InternalLengthToEndPlug)$$

$$InternalVolumeEachFilterCartridge_{max} = 48.686 \text{ in}^3$$

$$NumberOfCartridges := 4$$

$$TotalInternalVolume_{max} := (NumberOfCartridges) \cdot (InternalVolumeEachFilterCartridge_{max})$$

$$TotalInternalVolume_{max} = 194.743 \text{ in}^3$$

Bounding Limit - used to check experimental results. Measured water volume cannot exceed the volume of the filter cartridges external dimensions.

Figure 4B-1. First Article Filter Assembly Test. (6 sheets)

Next, the internal void space of the 4 filter cartridges is determined by simple experiment.

The empty, dry, filter assembly (with 4 cartridges) is weighed. Then, the filter assembly is submerged underwater. It is held louvered-end up, which allows all of the air to escape as bubbles.

After the air bubble release has stopped (about 5 minutes), the assembly is removed from the water. Water clinging to the exterior surfaces is gently towed off. The water-filled filter cartridge assembly is then placed on the scale and weighed.

Difference in the pre- and post-submergence weight is used to compute the volume of water retained in the filter cartridges. This retained water volume is equivalent to the void volume of a dry filter assembly.

WaterDensity := 0.9991 grams/cubic centimeter

FullWeight := 10780 bracket plus 4 filter cartridges plus water, grams

EmptyWeight := 8073 bracket plus 4 filter cartridges, grams

WaterWeight := (FullWeight - EmptyWeight)

WaterWeight = $2.707 \cdot 10^3$ grams

WaterVolume := $\frac{\text{WaterWeight}}{\text{WaterDensity}}$ cubic centimeters

WaterVolume = $2.709 \cdot 10^3$ cubic centimeters

To go from cubic centimeters unit to cubic inches we use the conversion factor of 0.061:

WaterVolume .061 = 165.276 cubic inches - This is equivalent to the total void volume inside all four cylinders of a dry MCO filter assembly,

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APPENDIX 4C

**SUMMARY OF VERIFICATION AND VALIDATION FOR
HANSF AND SINDA/FLUENT COMPUTER CODES**

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LIST OF TERMS

CVDF	Cold Vacuum Drying Facility
MCO	multi-canister overpack
SNF	spent nuclear fuel
TGA	thermo-gravimetric analysis
V&V	verification and validation

APPENDIX 4C

**SUMMARY OF VERIFICATION AND VALIDATION FOR
HANSF AND SINDA/FLUINT COMPUTER CODES**

Verification and validation (V&V) of the HANSF and SINDA/FLUINT thermal codes and models used for the safety basis analysis of the multi-canister overpack (MCO) and transportation cask were accomplished through a combination of independent review of the modeling assumptions, physical tests, thermal test cases to verify the basic modeling functions of the codes, independent review of the models and results obtained from them, and cross comparison with independently developed thermal models. The following sections provide an overview of these activities for both codes.

4C.1 INDEPENDENT REVIEW OF MODELING ASSUMPTIONS

HNF-SD-SNF-TI-015, *Spent Nuclear Fuel Project Technical Databook* (Reilly 1998), provides many of the key assumptions and much of the data used by both the HANSF and SINDA/FLUINT thermal models. Given the importance of this information, the critical underlying documents to the databook were subjected to independent review by recognized experts not associated with the Spent Nuclear Fuel (SNF) Project. Abridged resumes detailing the qualifications of each reviewer are provided in the attachment to this appendix. The independent reviews covered such critical assumptions as the MCO water content, gas generation from radiolytic water dissociation, and the particulate water content and behavior.

Table 4C-1 provides a the name of each independent reviewer; a summary of the reviewer's credentials; the documents reviewed; and the Engineering Data Transmittal number assigned to the file that contains the document review plan, the reviewer's comments, the agreed-upon resolution to the comments, and the review checklist and final document approval provided by each reviewer. In each independent review, the reviewers examined and approved for use by the SNF Project such items such as the consistency and accuracy of the data, appropriateness of the assumptions, mathematical derivations, validity of the models or methodology, and consistency of the conclusions with the analytical results and data presented in each document in question. The reader is referred to the Engineering Data Transmittal files for the specifics of each independent review because the number of reviews and the documentation associated with each is too extensive for a full inclusion in this V&V overview. For convenience, Table 4C-2 presents a summary of the general findings and conclusions arranged by reviewer and by document.

In addition to the independent reviews presented in Table 4C-1, internal peer reviews are conducted as part of the normal release process for each project document to ensure the accuracy, consistency, and appropriateness of the data and assumptions presented. Together, these two sets of reviews provide a record of quality assurance for the various documents used in the development of the HANSF and SINDA/FLUINT thermal models.

Table 4C-1. Technical Baseline Document Reviewers.

Reviewer	Credentials	Documents reviewed	EDT
Dr. G. Apostolakis	Ph.D in Engineering Science and Applied Mathematics, California Institute of Technology, 1973 Currently on faculty at MIT	HNF-2234, <i>Sensitivity of Probabilistic MCO Water Content Estimates to Key Assumptions</i> (Duncan 1999a)	624308
Dr. N.E. Bibler	Ph.D in Radiation Chemistry, Ohio State University, 1965 Currently Senior Advisory Scientist at Savannah River Site	HNF-SD-SNF-CN-006, <i>Evaluation of Radiolytic Gas Generation from Water Dissociation in a Multi-Canister Overpack</i> (Pajunen 1997)	625099
Dr. K. Czerwinski	Ph.D in Nuclear Chemistry, University of California, Berkeley, 1992 Currently on faculty at MIT	HNF-2234, <i>Sensitivity of Probabilistic MCO Water Content Estimates to Key Assumptions</i> (Duncan 1999a)	624308
Dr. J.R. Divine	Ph.D. in Chemical Engineering, Oregon State University Currently Chief Engineer, ChemMet, Ltd	HNF-1523, <i>K Basins Particulate Water Content, Behavior, and Impact</i> (Duncan 1997)	623980
		HNF-1527, <i>Estimates of Particulate Mass in Multi-Canister Overpacks</i> (Sloughter 1999)	623980
		HNF-SD-SNF-CN-017, <i>Fuel Surface Area</i> (Rev. 2) (Duncan 1998)	625210
Dr. X. Feng	Ph.D. in Inorganic/Physical Chemistry, Catholic University, Washington, D.C. Currently at Ferro Corp., Independence, Ohio (formerly staff scientist at Pacific Northwest National Laboratory)	HNF-SD-SNF-TI-040, <i>MCO Internal Gas Composition and Pressure During Interim Storage</i> (Rev. 1) (Frederickson 1998)	624307
Mr. G. Maurin	Graduate Engineer in Nuclear Engineering, Ecole Polytechnique, Paris Currently Manager, Technology Department, SGN	HNF-1523, <i>K Basins Particulate Water Content, Behavior, and Impact</i> (Duncan 1997)	623980
		HNF-1527, <i>Estimates of Particulate Mass in Multi-Canister Overpacks</i> (Sloughter 1999)	623980
		HNF-2256, <i>Simulation of Normal and Off-Normal Multi-Canister Overpack Behavior</i> (Duncan 1999b)	625176
Dr. D.R. Olander	ScD. in Chemical Engineering, MIT. Currently on faculty at University of California, Berkeley	HNF-1523, <i>K Basins Particulate Water Content, Behavior, and Impact</i> (Duncan 1997)	623980
		HNF-1527, <i>Estimates of Particulate Mass in Multi-Canister Overpacks</i> (Sloughter 1999)	623980
		HNF-1851, <i>Cold Vacuum Drying Residual Free Water Test Description</i> , Rev.1 (Pajunen 1998)	625201

EDT = Engineering Data Transmittal (number).

Table 4C-2. Summary of Reviewers' Findings and Conclusions. (2 sheets)

Reviewer	Documents reviewed	Findings/conclusions
Dr. G. Apostolakis	HNF-2234, <i>Sensitivity of Probabilistic MCO Water Content Estimates to Key Assumptions</i> (Duncan 1999a)	Seven comments were issued against the report. The issued report was found to be responsive to all of the comments.
Dr. N.E. Bibler	HNF-SD-SNF-CN-006, <i>Evaluation of Radiolytic Gas Generation from Water Dissociation in a Multi-Canister Overpack</i> (Pajunen 1997)	One comment issued against the report. The issued report was found to represent a thorough search of the literature and the bounding acceleration values are appropriate.
Dr. K. Czerwinski	HNF-2234, <i>Sensitivity of Probabilistic MCO Water Content Estimates to Key Assumptions</i> (Duncan 1999a)	Reviewed report with Dr. G. Apostolakis. All comments issued against the report were found to be incorporated in the issued report.
Dr. J.R. Divine	HNF-1523, <i>K Basins Particulate Water Content, Behavior, and Impact</i> (Duncan 1997)	Fifteen editorial and minor calculation-related comments issued against the report. Generally, the assumptions and calculations made in the report were found to be reasonable and correct. The reviewer did feel that the assumptions were generally overly conservative.
	HNF-1527, <i>Estimates of Particulate Mass in Multi-Canister Overpacks</i> (Slougher 1999)	Six editorial comments issued against the report. Conclusions drawn in the report were found to be reasonable based on available information. Further evaluations were suggested for statistical purposes, but the reviewer did not feel it warranted holding up the project.
	HNF-SD-SNF-CN-017, <i>Fuel Surface Area</i> (Rev. 2) (Duncan 1998)	Twenty comments initially issued were found to have been incorporated in the report and the report was found to be technically acceptable. The application and conclusions are satisfactory and sound. Minor editorial comments issued regarding improvement in material presentation.
Dr. X. Feng	HNF-SD-SNF-TI-040, <i>MCO Internal Gas Composition and Pressure During Interim Storage</i> (Rev. 1) (Frederickson 1998)	Twenty-seven comments issued against the review copy. The revised report was found to have either incorporated the comments or responded to the reviewer's satisfaction. The issued report was found to be technically acceptable.

Table 4C-2. Summary of Reviewers' Findings and Conclusions. (2 sheets)

Reviewer	Documents reviewed	Findings/conclusions
Mr. G. Maurin	HNF-1523, <i>K Basins Particulate Water Content, Behavior, and Impact</i> (Duncan 1997)	Approximately 21 comments or findings were issued for the report. The conclusion was that it is technically feasible to justify the water content in the MCO after cold vacuum drying processing, the exact amount of free water contained in the particulate would be better supported with more data.
	HNF-1527, <i>Estimates of Particulate Mass in Multi-Canister Overpacks</i> (Slougher 1999)	Approximately 18 comments or findings were issued for the report. The conclusions are addressed by the statement above.
	HNF-2256, <i>Simulation of Normal and Off-Normal Multi-Canister Overpack Behavior</i> (Duncan 1999b)	Approximately 26 comments or findings were issued for the report in letter form and 5 in RCR format. The RCR comments were addressed to the satisfaction of the reviewer.
Dr. D.R. Olander	HNF-1523, <i>K Basins Particulate Water Content, Behavior, and Impact</i> (Duncan 1997)	Comments were issued for the report in 7 general areas. The water in the generated particulate was highlighted as an area of uncertainty within the report, as was the formation of uranium hydrides. The use of a geochemical code like EQ3 and the shortening of the document were suggested.
	HNF-1527, <i>Estimates of Particulate Mass in Multi-Canister Overpacks</i> (Slougher 1999)	Comments were related to correcting equation units, the amount of sludge oxides, and a request for a brief description of the environment for each process step. Overall, the document was found to be a careful assessment of the quantities of nonmetallic materials that could retain water.
	HNF-1851, <i>Cold Vacuum Drying Residual Free Water Test Description</i> , Rev. 1 (Pajunen 1998)	Eight review comments were issued relating to model assumptions and approaches. All comments were resolved to the reviewer's satisfaction.

4C.2 PHYSICAL TESTING

The physical testing accomplished to date in direct support of the SNF Project falls into two categories: (1) testing to support the assumptions regarding the reactivity of irradiated N Reactor fuel to various environments and (2) the testing of the equipment and procedures to be used in the Cold Vacuum Drying Facility (CVDF) to remove the 'free' water remaining in an MCO after draining. HNF-4206, *Oxidation Kinetics of K Basin Fuel* (Trimble 1999), and HNF-SD-SNF-CN-019, *Reaction Rate Constant for Dry Air Oxidation of K Basin Fuel* (Trimble 1998) present the results of thermo-gravimetric analysis (TGA) testing of N Reactor fuel samples in dry air, moist air, and moist helium atmosphere. As depicted in Figure 4C-1, the

chemical reaction rates deduced from the TGA testing on damaged and unirradiated K Basins fuel in a moist, inert gas are bounded by the literature rate data (i.e., Pearce 1989) assumed in HNF-SD-SNF-TI-015 (Reilly 1998).

Figure 4C-1 illustrates that the test data for the N Reactor fuel (symbolized by the open circles and the solid triangular data points in the figure) are encompassed by the literature-based rate equation incorporated within the HANSF, SINDA/FLUINT, and GOTH_SNF¹ thermal models. As such, the figure demonstrates that the assumed chemical reaction rate equation used in the models has been validated against both the level and trend in the reaction rate test data seen for the N Reactor fuel.

The TGA results, plus data for the drying test on whole elements and the observed gas generation during shipping of fuel assemblies, were combined in HNF-4165, *Uranium Oxidation Rate Summary for the Spent Nuclear Fuel Project* (Pajunen 1999), to confirm that the chemical reaction rate equations and composite surface area enhancement factors presented in HNF-SD-SNF-TI-015 (Reilly 1998) were conservative and bounding. Figure 4C-2 illustrates the level of conservatism produced by the combination of the literature-based reaction rates and the composite enhancement factors. This safety margin would suggest that even if the logic used in the HANSF and SINDA/FLUINT thermal models does not capture every nuance in the actual chemical reaction rate, sufficient margin exists to prevent the codes from underestimating the true level of the reaction rate. Such validation of a model's conservatism is an important issue in software V&V when it relates to a code used to compute the safety basis of a design and/or process.

Cold vacuum drying tests were conducted in 1998 and early 1999 using a 'first article' MCO, fuel and scrap baskets, and CVDF skid-mounted equipment. The spent fuel payload was simulated using carbon steel tubing for the fuel assemblies and crushed carbon steel pipe for the scrap. Water-holding cavities were formed at the tops of selected simulated fuel assemblies to mimic damaged fuel assemblies. The results of the testing and the post-test analytical simulations of the tests, presented in SNF-4083, *Hanford Spent Nuclear Fuel Cold Vacuum Drying Post-Process Test GOTH_SNF Model Simulations of Proof of Performance of First Article Tests and Equipment Performance Evaluation* (Crea and Fryer 1999), demonstrated that the CVDF equipment and processes would remove the free water remaining after MCO draining within the time allotted in the CVDF process plan. Further, the close match between the actual and analytically predicted drying times demonstrated that the GOTH_SNF thermal model used for this work accurately modeled the heat transfer characteristics of the MCO, the fuel baskets, and the simulated spent fuel payload. Section 4C.5 presents a summary of the match between the first article drying test results and the analytical predictions using the GOTH_SNF code.

¹GOTH_SNF developed by John Marvin, Incorporated, Richland, Washington. GOTH_SNF is a derivative code to GOTHIC, licensed by the Electric Power Research Institute, Palo Alto, California.

The GOTH_SNF code is a derivative of the COBRA-TF code, a long established thermal-hydraulics code used for nuclear reactor safety analysis. The specific GOTH_SNF models used for the SNF Project have been internally reviewed by the contractor (John Marvin, Inc.) under its quality assurance program. John Marvin, Inc., is on the Project Hanford Management Contract evaluated supplier listing. An independent peer review was not conducted of this thermal model because it is not being used for safety basis calculations. However, the peer review of the basic assumptions used in the HANSF and SINDA/FLUENT thermal models also applies to the basic assumptions used in the GOTH_SNF analysis.

4C.3 VERIFICATION AND VALIDATION OF HANSF COMPUTER CODE AND MODEL

The HANSF computer code² is a generalized fluid and aerosol model that uses 'tank' and 'tube' elements to simulate a specific problem. For use on this project, the HANSF model incorporated MCO model coding developed specifically for the SNF Project. Thus its modeling capabilities and functionality are tailored to the specific MCO design and thermal problems encountered. The contents of the MCO are modeled using internal control volumes, with separate control volumes used to represent the individual scrap and fuel baskets. Within each control volume, the model accounts for the chemical reactions, tracks water content and species concentrations, and simulates the heat transfer mechanisms. Junctions are used to connect the MCO internal control volumes and allow forced and countercurrent flow between the scrap and fuel basket regions. Upper and lower control volumes are used to represent plenum spaces above and below the scrap and fuel baskets.

SNF-3650, *HANSF 1.3 User's Manual* (Plys et al. 1999) provides the V&V basis for the code. As presented in the HANSF User's Manual (Plys et al. 1999), the V&V for HANSF consists of three parts.

- The experimental basis supporting many of the modeling assumptions, such as uranium oxidation, hydrate decomposition, and aerosol transport and deposition, was established and its application to the MCO analysis presented.
- Separate effects testing was conducted using the code to validate the correct function of the model for individual phenomena. Seventeen separate effects tests were conducted and are documented within the HANSF User's Manual (Plys et al. 1999).
- Four integral tests for typical MCO transients demonstrate the model-predicted performance is reasonable and consistent with expectations.

²The HANSF code was developed by Fauske & Associates, Incorporated, Burr Ridge, Illinois.

In addition to the V&V basis presented in the HANSF User's Manual (Plys et al. 1999), the HANSF code has been validated by comparison with independent results obtained using the SINDA/FLUINT and GOTH_SNF codes. The following sections provide an overview of V&V activities in each of the three areas presented in the HANSF User's Manual (Plys et al. 1999) and the comparison with independent results.

4C.3.1 VERIFICATION AND VALIDATION OF HANSF COMPUTER CODE BASED ON THE EXPERIMENTAL BASIS

The experimental basis for the HANSF code includes literature-based results for oxidation rates, ignition theory, hydrate decomposition, aerosol transport and deposition, countercurrent flow, and natural circulation within the MCO. These documented experimental results are used to provide the technical basis for the model function in these areas.

The experimental database for oxidation rates of uranium under various environments is the same as that confirmed and extended under the physical testing described in Section 4C.2. As such, the basis for the HANSF code in this area has been validated by experimental results conducted on the specific N Reactor fuel to be handled in the MCO. While the HANSF User's Manual (Plys et al. 1999) also presents data for the ignition theory of powdered uranium, this theoretical development is never actually used in the model. Instead, it is simply presented as a demonstration of the necessity of computing the temperature gradient across the MCO scrap and fuel baskets and for preventing excessive temperatures within the spent fuel payload.

The decomposition of unirradiated $\text{UO}_3 \cdot 2\text{H}_2\text{O}$ is based on experimental work from seven TGA runs conducted by Pacific Northwest National Laboratory on N Reactor spent fuel (Abrefah et al. 1997a, Abrefah et al. 1997b). These tests consisted of a series of five heatup steps from ambient temperature to 625 °C (i.e., 50 °C, 75 °C, 300 °C, 425 °C, and 625 °C) followed by dwell times of 6 to 12 hours after each heatup step. A 1 °C per minute heatup rate was generally used, except for the final step from 425 °C to 625 °C, which used a 2 °C per minute heatup rate. In addition, four runs (numbers 43 through 46) used a 0.2 °C per minute heatup rate from 75 °C to 300 °C. The weight loss of the sample was measured over each heatup step and dwell time period. The results of these experiments were used to develop a kinetic rate law that provides the decomposition rate as a function of temperature and pressure. The literature work of Danroc (1982) provides further background information on the decomposition of unirradiated $\text{UO}_3 \cdot 2\text{H}_2\text{O}$. The experimental results also relate the percent weight loss (i.e., decomposition) as a function of time and temperature. While the derived rate law for the decomposition of $\text{UO}_3 \cdot 2\text{H}_2\text{O}$ is used to conservatively estimate the amount of reactants available to drive the chemical reactions and gas generation during the cold vacuum drying process and dry transportation, no credit is taken for the potential reduction in total MCO water content caused by any $\text{UO}_3 \cdot 2\text{H}_2\text{O}$ decomposition during cold vacuum drying.

The aerosol transport and deposition function within the HANSF code is based on an exact, numerical solution to the aerosol equation (Epstein and Ellison 1988). The AB-5 experiment performed at the Hanford Site (Hilliard et al. 1983) provides the validation basis for the time-

dependant calculations contained in HANSF for aerosol sources, leakage through model junctions, and removal mechanisms such as impaction and sedimentation.

Because countercurrent flow may be an important mechanism in MCO accident scenarios, the gas and aerosol transport caused by countercurrent flow is accounted for within the HANSF code. This calculation is based on the correlations of countercurrent flow as a function of density differences (Epstein 1988).

Experimental results for convection flows within enclosures with high aspect ratios are used to dismiss the potential contribution of convection to the heat transfer between fuel assemblies and within the fuel assembly annulus. The same conclusion is reached for the porous layers within the scrap basket. Ignoring the potential presence of convection flows within the MCO is conservative.

4C.3.2 VERIFICATION AND VALIDATION OF THE HANSF COMPUTER CODE BASED ON SEPARATE EFFECTS TESTING

The function of the HANSF code for correctly predicting individual phenomena pertinent to the thermal-hydraulic modeling of the MCO was validated for the following 17 separate effects (Plys et al. 1999):

1. Region thermodynamics
2. Fog formation
3. Intercompartmental flow
4. Heat transfer modeling
5. Aerosol transport and deposition
6. Countercurrent flow
7. Aerosol deentrainment
8. Evaporation
9. Hydride formation and decomposition
10. Uranium oxidation
11. Entrainment
12. Hydrate decomposition
13. Fuel and scrap basket heat transfer
14. Nitriding
15. Vacuum and recirculation pump
16. Radiation networks
17. Overall energy balance.

This section presents a selection of the special effects testing conducted with the HANSF code to illustrate the type of testing conducted and agreement achieved with the known value. For a full discussion of the testing for the 17 separate effects, the reader is referred to Chapter 7 of the HANSF User's Manual (Plys et al. 1999).

The ability of the code to correctly compute the energy state of the various control volumes was validated by initiating the control volumes at a reference energy state and then adding an energy source. The computed energy state was then compared to steam table data to verify that the correct new energy state was determined. Table 7-1 of the HANSF User's Manual (Plys et al. 1999), reproduced in Figure 4C-3, demonstrates that very good agreement was achieved on a large temperature range. Likewise, Table 7-2 of the HANSF User's Manual (Plys et al. 1999), reproduced in Figure 4C-4, demonstrates that the formation or depletion of aerosols (i.e., fog formation) is correctly reflected in the computed equilibrium state.

Heat transfer modeling was validated using the classical problem of a slab at a uniform initial temperature suddenly exposed to a constant temperature fluid at one end while the other end is kept insulated. A constant heat transfer coefficient was used. The analytical solution for the initial 50,000 seconds of the transient is taken from *Fundamentals of Heat Transfer* (Incropera and DeWitt 1981). Table 7-4 from the HANSF User's Manual (Plys et al. 1999), reproduced in Figure 4C-5, demonstrates that good agreement is achieved between the HANSF code and the analytical results.

The validation of the code-implemented uranium oxidation rates was achieved by exercising the code with a specified MCO gas condition and confirming that the computed oxidation rate matched the hand calculations. Four sets of MCO gas conditions were used:

1. 75 °C, 1% steam, 20% oxygen, and 79% nitrogen
2. 300 °C, 1% steam, 20% oxygen, and 79% nitrogen
3. 75 °C, 51% oxygen, and 49% nitrogen
4. 300 °C, 51% steam, and 49% nitrogen.

The enclosed MCO consisted of three regions with a total volume of 0.56 m³, and a total effective oxidation area of 128 m². No decay heat, evaporation, condensation, or gas heat transfer models were in effect. Cases with 75 °C were run for 1,000 seconds while those with 300 °C were run for 1 second. Table 7-6 from the HANSF User's Manual (Plys et al. 1999), reproduced in Figure 4C-6, demonstrates the good agreement between the hand- and code-computed reaction rates.

Again, for a complete discussion of the separate effects testing conducted for the four special effects discussed above and the 13 other special effects, see the Chapter 7 of the HANSF User's Manual (Plys et al. 1999).

4C.3.3 VERIFICATION AND VALIDATION OF HANSF COMPUTER CODE BASED ON INTEGRAL TESTING

The integrated operation of the HANSF code was validated using four tests cases for typical MCO transients to demonstrate that the code computed reasonable performance consistent with expectations. The integral test cases were (1) simulation of a nominal cold vacuum drying

process, (2) the simulation of a shipping transient, (3) interim storage, and (4) cold vacuum drying with axial variation in the location of the fuel damage.

While exact analytical or test data are unavailable for direct comparison, the test cases do serve the function of verifying the integrated operation of the code with inputs that simulate expected operating conditions. The important conclusion drawn from the integral tests is that the results are consistent across the various control volumes, that the transient behavior is consistent with the model inputs for the heat generation, heat transfer, and decomposition rates, and that mass energy is conserved in the multivolume representation of the MCO.

4C.3.4 VERIFICATION AND VALIDATION OF HANSF COMPUTER CODE BASED ON INDEPENDENT MODELING

The final V&V activity conducted with the HANSF code was a comparison with the thermal results obtained from an independent code (i.e., GOTH_SNF). This activity is documented in SNF-5226, *Comparison Cases Simulated with HANSF 1.3.2 that Supplement the Thermal Analyses Documented in HNF-SD-SNF-CN-023* (Piepho 1999). The selected comparisons with the GOTH_SNF code are for two safety basis cases for CVDF operations.

The first comparison is for the case of normal vacuum drying with helium purge (i.e., 30 ft³/min maximum vacuum pumping with 0.7 ft³/min helium purge), assuming two scrap baskets and the normal amount of free water after MCO draining (i.e., 26.5 kg for the two-scrap-basket configuration). Figure 4C-7 presents a comparison of the predicted water vapor pressure at the top of the MCO (i.e., the top scrap basket for HANSF and the outlet for GOTH_SNF). This parameter was chosen as a summary basis for the comparison because its correct calculation requires an accurate determination of a number of thermal mechanisms within the MCO. These mechanisms include the combined heat transfer mechanisms and internal heat generation within the baskets required to correctly compute the temperature of the fuel assemblies and scrap basket and the evaporation rate of the water, and the mass flow balance mechanisms required to compute the MCO partial pressures throughout the MCO.

As seen from the figure, a good agreement was achieved between the two codes for both the total pressure within the MCO and the partial pressure of the water vapor. The HANSF results are seen as being conservative in comparison with the GOTH_SNF results because their slightly lower predicted water evaporation rate (as evidenced by the lower partial pressure of water vapor) yields a longer period of time during which the damaged fuel sections are subjected to chemical reaction with the water vapor. As a direct result of the lower predicted evaporation rate, the predicted MCO basket dryout time under a continuous vacuum drying cycle is 2 to 3 hours longer under the HANSF analysis than under the GOTH_SNF analysis (based on extrapolation of the GOTH_SNF results).

Plots of additional parameters from the HANSF and GOTH_SNF results for this case are presented in Figures 4C-8 and 4C-9, respectively. Figure 4C-8 is a partial reproduction of Figure A-3 from SNF-5226, while the GOTH_SNF results presented in Figure 4C-9 are taken

from Figures 3.31 to 3.35 in JMI980305-10, *GOTH_SNF Simulations of Selected CVDF Vacuum Pumping and Design Basis Accident Cases* (Thurgood 1999). As noted from Figure 4C-9, the GOTH_SNF transient analysis was stopped after 10 hours. This end time was arbitrarily selected to limit computer time because the maximum time permitted under normal vacuum drying conditions by the CVDF process specification is 8 hours.

Annotations are provided on these figures to aid the understanding of the plotted information and to further demonstrate the good match between the predicted results between the two codes. Specifically, Figure 4C-8 demonstrates that the scrap basket temperatures predicted by the HANSF model after 10 hours under normal vacuum drying conditions are within 6 °C of the GOTH_SNF model temperatures and within 2 °C of the GOTH_SNF temperatures for the center intact fuel basket. The bottom scrap basket runs cooler during the initial portion of the transient because its 6 kg of water (as opposed to the 1.5 kg assumed for the top scrap basket) results in an extended period under evaporative cooling. This uneven water distribution was assumed in order to evaluate the maximum temperatures within a scrap basket that dries out prior to the removal of the bulk of the free water within the MCO. In addition to achieving a good agreement in temperature levels, comparing the plots presented in Figures 4C-8 and 4C-9 shows that the predicted temperature trend lines are very similar between the models.

The second comparison case is for a degraded vacuum drying cycle (i.e., without the helium purge). This case was selected to evaluate the thermal model's ability to evaluate the impact on thermal performance of the absence of the helium purge gas. Again two scrap baskets and 26.5 kg of free water after MCO draining are assumed. Figure 4C-10 presents a comparison of the predicted water vapor pressure at the top of the MCO. Total pressure is not shown on this plot because without helium purge and the relatively low hydrogen generation rate, the plot of water vapor pressure and total MCO pressure would essentially be the same line.

Again, as seen from the figure, excellent agreement was achieved between the two codes for the partial pressure of the water vapor for the majority of the transient. As with the case for normal vacuum drying, the HANSF results provided a slightly more conservative estimate of the drying performance within the MCO. Plots of additional parameters from the HANSF and GOTH_SNF results for this case are presented in Figures 4C-11 and 4C-12, respectively. Figure 4C-11 is a reproduction of Figure A-1 from SNF-5226, while the GOTH_SNF results in Figure 4C-12 are taken from Figures 3.2 to 3.6 in JMI980305-10. Annotations provided on Figure 4C-11 demonstrate that the temperature match achieved between these two codes is similar to that seen in the previous example, while a comparison of the plots in Figures 4C-11 and 4C-12 again demonstrates the two codes predict similar temperature trends for scrap and intact fuel baskets.

A comparison of results obtained using the HANSF code with those obtained using the SINDA/FLUINT code is presented and discussed in Section 4.4.3.3 of Chapter 4.0 for dry transportation. A comparison with results from the SINDA/FLUINT code serves as an additional means of validating the HANSF code by showing that its results compare favorably with those from a second, independently developed thermal model. Figure 4-25 presents a comparison between the HANSF analysis case 'SHIPC1', as taken from Appendix T of HNF-2256, *Simulation*

of Normal and Off-Normal Multi-Canister Overpack Behavior (Plys and Duncan 1998), and the SINDA/FLUINT model. This analysis case assumes one scrap basket with 45,000 cm² of reaction area and four fuel baskets with 425 cm² of reaction area each. A composite reaction enhancement factor of 22 is used. As such, the analyzed case represents an MCO configuration with one safety basis scrap basket and four design basis baskets of intact fuel.

As seen from Figure 4-25, sheet 2, the HANSF analysis predicts a maximum fuel basket temperature of approximately 390 K, or an increase of 50 K after 75 hours ([500,000 minus 230,000 seconds]/3,600 seconds per hour). In comparison, the results of the SINDA/FLUINT analysis, illustrated in Figure 4-25, sheet 1, indicate a peak temperature of 110 °C (380 K) and an increase of 45 °C over the 75-hour period following the 8-hour hold at the CVDF. The higher temperature increase attributed to HANSF is in part due to the fact that the HANSF analysis assumes a severe ambient range of 95 °F to 115 °F, while the SINDA/FLUINT analysis assumes a typical Hanford July day cycle of 74 °F to 115 °F. The cooler night temperatures assumed by the SINDA/FLUINT model slow the cask heatup and, thus, the maximum component temperatures noted within the canister.

Further, the HANSF code predicts a maximum canister pressure of 285 kPa or 41 lb/in² absolute and a pressure increase of 13 lb/in² ([285 minus 195 kPa]/6,894.76 kPa/lb/in² absolute) after 75 hours versus a 16 lb/in² pressure increase estimated by the SINDA/FLUINT results. The higher canister pressure rise computed by the SINDA/FLUINT model is primarily related to slight differences in the assumed release rate for the free and chemically bound water.

Taken together, these comparative results from independent thermal models establish the general validation of the HANSF thermal model for cold vacuum drying and dry transportation operations.

4C.4 VERIFICATION AND VALIDATION OF SINDA/FLUINT COMPUTER CODE AND MODEL

The analytical thermal model for the cask-MCO was developed using the SINDA/FLUINT heat transfer code (SINDA/FLUINT 1998). This finite-difference, lumped-parameter code, developed under the sponsorship of the National Aeronautics and Space Administration, Johnson Space Center, has been evaluated and validated in a government-sponsored study against several other thermal codes for simulating the thermal response of transportation packages (Glass et al. 1988). The SINDA/FLUINT code has previously been used for the analysis of several transportation casks used for nuclear material under both Department of Energy and Nuclear Regulatory Commission review, including most recently the radioisotope thermoelectric generator transportation package (Ferrell 1998).

The SINDA/FLUINT code provides the capability to simulate steady-state and transient temperatures using temperature-dependent material properties and heat transfer via conduction,

convection, and radiation (SINDA/FLUINT 1998). Complex algorithms may be programmed into the solution process for the purposes of computing the various heat transfer coefficients as a function of the local geometry, gas thermal properties, and temperatures, or for example, to estimate the effects of buoyancy-driven heat transfer.

A major feature of the SINDA/FLUINT code employed for this modeling is the ability to use submodels to represent sections of the cask, fuel baskets, or payload having common geometry and to connect the individual submodels to form a complete model of the MCO and cask. This approach simplifies the modeling and verification process by minimizing the amount of original coding required to provide a complete thermal representation of the system.

4C.4.1 VERIFICATION AND VALIDATION OF SINDA/FLUINT COMPUTER CODE BASED ON TEST CASES

The basic performance of SINDA/FLUINT code for handling heat transfer problems was evaluated for conduction, convection, and radiation heat transfer under both transient and steady-state conditions. These modes of heat transfer and conditions are directly applicable to the intended use of the computer program for performance of thermal analysis. The following four specific analytical cases were examined:

1. The transient behavior of a semi-infinite slab subject to a convection boundary condition — The classical, closed-form solution for this problem (Ozisik 1980) was used as the basis of comparison.
2. The steady-state behavior of a composite cylinder, with uniform heat generation within an inner cylinder, conduction through a surrounding cylinder, and a convection boundary condition at the outer radius — The documented results of a benchmark study by Sandia National Laboratories (Glass et al. 1988) were used as the basis of comparison.
3. The steady-state and transient behavior of a cask containing spent nuclear fuel and involving internal heat generation, conduction and radiation heat transfer — The documented results of a benchmark study by Sandia National Laboratories (Glass et al. 1988) were used as the basis of comparison.
4. Sample problem contained in the input file '*sinsamp.inp*', which was provided with the installation software and the output results as documented in the SINDA/FLUINT user's manual (SINDA/FLUINT 1998)— Validation was made based on direct comparison of the output files.

4C4.1.1 Validation Problem 1 — The Transient Behavior of a Semi-Infinite Slab Subject to a Convection Boundary Condition

Figure 4C-13 shows a schematic diagram of the problem along with a list of the problem parameters. The slab, which is initially at a uniform temperature of 0 °C, is subjected to a convection boundary condition with an environmental temperature of 100 °C. The semi-infinite slab is modeled in SINDA/FLUINT using 100 nodes, 0.05-m apart, for a total length of 5 m. Figure 4C-14 shows a plot of the non-dimensionalized temperature profile through the slab at time increments of 6, 12, 18, and 24 hours for the classical, closed form (Ozisik 1980) and the SINDA/FLUINT solutions. The non-dimensionalized temperature is defined as $(T - T_i)/(T_{inf} - T_i)$, where T is the temperature at the current time and location in the slab, T_i is the initial temperature of the slab, and T_{inf} is the boundary temperature of the external environment.

Conclusion: The SINDA/FLUINT results match the closed-form solution. Computer system input and output files are included in Appendix A of QML 1000.001, *Computer Verification For SINDA/FLUINT, Version 4.0* (Q-Metrics 1999).

4C4.1.2 Validation Problem 2 — The Steady-State Behavior of a Composite Cylinder with Heat Generation and Convection

Figure 4C-15 shows a schematic diagram of verification problem 2, and Table 4C-3 lists the problem parameters. The domain consists of an inner cylinder (Region I) with uniform heat generation, surrounded by an outer cylinder (Region II) with no heat generation. A convective boundary condition exists at the outer radius of the outer cylinder. The problem was modeled axisymmetrically in SINDA using 5 nodes for the inner cylinder and 15 nodes for the outer cylinder. Table 4C-4 compares the SINDA/FLUINT results with the temperatures reported in the Sandia National Laboratories benchmark study (Glass et al. 1988).

Table 4C-3. Problem Parameters for the Composite Cylinder of Verification Problem Two.

Region	Inner/outer radius (m)	ρ (kg/m ³)	k (W/m °C)	C_p (J/kg °C)	g (W/m ³)
I	0.0 / 0.274	16.02	69.2	4,187	11,090
II	0.274 / 0.914	16.02	34.6	4,187	0
Boundary conditions		$h = 5.67 \text{ W/m}^2 \text{ °C}$, $T_{inf} = 54.4 \text{ °C}$			

Table 4C-4. Benchmark and SINDA/FLUINT Temperature Results for Verification Problem Two.

	Centerline (°C)	Interface (°C)	Outer Edge (°C)
Benchmark	152	149	135
SINDA/FLUINT	153.1	148.8	135.0

Note: Benchmark results are from R. E. Glass, et al., 1988, *Standard Thermal Problem Set for the Evaluation of Heat Transfer Codes Used in the Assessment of Transportation Packages*, SAND88-0380, Sandia National Laboratory, Albuquerque, New Mexico.

Conclusion: The SINDA/FLUINT results match the results from the benchmark study (Glass et al. 1988). Computer system input and output files are included in Appendix A of QML1000.001 (Q-Metrics 1999).

4C4.1.3 Validation Problem 3 — The Steady-State and Transient Behavior of a Spent Fuel Cask Subject To Radiant Heat Transfer Conditions

Figure 4C-16 shows a schematic diagram of verification problem 3, and Table 4C-5 lists the problem parameters. The cask consists of an inner cylinder with uniform heat generation (Region I), surrounded by a cylinder of steel (Region II), a voided neutron shield (Region III), and the outer steel wall of the neutron shield (Region IV). The single mode of heat transfer between Regions II and IV is thermal radiation. Another shield in the form of a plate is positioned below the cask. The cask-shield arrangement is assumed to transfer heat to the surrounding environment by thermal radiation only. There is also an exchange of thermal radiation between the cask and the shield.

The steady-state behavior of the cask is computed for an environmental temperature of 54.4 °C. The transient behavior is then computed assuming that the cask is subjected to a fire environment of 800 °C for 30 minutes followed by a return to the 54.4 °C environment.

Region I was modeled axisymmetrically in SINDA/FLUINT using five nodes in the radial direction. Region II was modeled in two dimensions with eight nodes radially and six nodes around the circumference (for a total of 48 nodes). Region III had no nodes because it was a nonparticipating medium allowing only radiation heat transfer. Region IV was modeled with one node radially and six nodes circumferentially (for a total of 6 nodes). The external shield was modeled with one node. Radiation conductors were used between Regions II and IV, between Region IV and the shield, and between Region IV and the surrounding environment. Appropriate view factors were calculated using the "crossed-string" method outlined in *Thermal Radiation Heat Transfer* (Siegel and Howell 1992).

Table 4C-5. Problem Parameters for the Composite Cylinder of Verification Problem Three.

Region	Inner/outer radius (m)	ρ (kg/m ³)	k (W/m °C)	Cp (J/kg °C)	g (W/m ³)
I	0.0 / 0.165	2,707	242	896	38,320
II	0.165 / 0.387	7,833	45.0	473	0
III	0.387/ 0.540	Nonparticipating medium with radiant heat exchange only			
IV	0.540/ 0.546	7,833	45.0	473	0
Shield	1.092 m long, 0.025 m thick, 0.305 m from cask	7,833	45.0	473	0
Boundary conditions		$T_i = 54.4\text{ °C}$ $T_{inf} = 54.4\text{ °C}$, (initial steady-state) $= 800\text{ °C}$, $0 < t < 30\text{ minutes}$ $= 54.4\text{ °C}$, $30 < t < 90\text{ minutes}$			
Radiation conditions		$\epsilon = \alpha = 1$ for all surfaces			

Table 4C-6 compares the SINDA/FLUINT results at various times and locations with the temperatures reported in the benchmark study (Glass et al. 1988). The SINDA/FLUINT results are well within the variations (i.e., +15 °C) reported in the benchmark study (Glass et al. 1988) between the various computer codes evaluated.

Conclusion: The SINDA/FLUINT results match the benchmark study (Glass et al. 1988). Computer system output files are included in Appendix A of QMI.1000.001 (Q-Metrics 1999).

4C4.1.4 Verification Problem 4 — Sample Problem '*sinsamp.inp*'

Verification problem 4 is sample problem '*sinsamp.inp*', as provided with the SINDA/FLUINT code. This problem demonstrates that the basic capabilities and options of the SINDA/FLUINT code, such as the use of array data and time-varying sources, are functioning as expected. As such, this problem serves to verify that the SINDA/FLUINT code produces the results the code developers had intended.

Table 4C-6. Benchmark and SINDA/FLUINT Temperature Results for Verification Problem Three.

Time (min)	T1 (°C)	T2 (°C)	T3 (°C)	T4 (°C)	T5 (°C)
0	88/88.2	146/147.1	217/216.9	139/138.6	207/207.5
30	765/765.7	664/661.5	261/260.4	689/686.7	350/345.4
90	205/205.2	242/248	314/312.9	203/202.5	301/303.8

Note: Benchmark results are from R. E. Glass, et al., 1988, *Standard Thermal Problem Set for the Evaluation of Heat Transfer Codes Used in the Assessment of Transportation Packages*, SAND88-0380, Sandia National Laboratory, Albuquerque, New Mexico.

Figure 4C-17 illustrates the schematic diagram of the sample problem provided with the SINDA/FLUINT installation program. A hollow cubic satellite is hovering centered (i.e., negligible conductive contact) within a hollow cubic cargo bay in low earth orbit. Within the satellite a small solid cube hovers. The cargo bay is initially in a planet-oriented equatorial orbit and is open on the side facing away from the planet. When the spacecraft passes behind the planet, the satellite is deployed in the same orbit and is reoriented to be sun-facing (see Figure 4C-17). The initial and final orbital-average conditions of the satellite are desired as well as a simulation of the deployment event. The relevant characteristics are as follows:

- Cargo bay (insulated on outside)
 - Number of panels 5
 - Overall size 2 ft x 2 ft x 2 ft
 - Panel thickness negligibly thin
 - Material Aluminum
- Satellite
 - Number of panels 5
 - Overall size 1 ft x 1 ft x 1 ft
 - Panel thickness 1/8 in.
 - Enclosed solid cube size 1 in. x 1 in. x 1 in.
 - Material Aluminum
- All active surfaces
 - IR emissivity 0.8
 - Solar absorptivity 0.2
 - Insulated at edges.

Figure 4C-18 illustrates the transient thermal performance as presented in the SINDA/FLUINT user's manual (SINDA/FLUINT 1998). Figure 4C-19 presents the transient thermal results as provided with the sample problem, and Table 4C-7 presents the results as determined by this installation of the SINDA/FLUINT program. Comparison of the two thermal summaries shows that this installation of SINDA/FLUINT provides results that match exactly with those provided with the benchmark sample problem.

Table 4C-7. SINDA/FLUINT-Generated Temperature Results for Verification Problem Four.

Problem Results From V&V Run								
SUMMARY TABLE OF SINDA SAMPLE PROBLEM: SIMPLIFIED SATELLITE DEPLOYMENT ++ TEMPERATURES ++								
TIME	CUBE	OUTSIDE	INSIDE	OTHER SIDE PANELS-----				STATUS SUMMARY
0.000	-17.238	-34.669	-3.3743	-19.260	-19.259	-14.284	-14.285	ORBITAL AVERAGE IN CARGO BAY
0.000	-17.238	-34.669	-3.3743	-19.260	-19.259	-14.284	-14.285	HALF ORBIT IN CARGO BAY
0.077	-16.750	-21.778	16.738	-8.2483	-8.2476	-2.3955	-4.5284	HALF ORBIT IN CARGO BAY
0.153	-15.462	-12.047	27.114	1.01318E-02	1.07422E-02	9.9414	1.8962	HALF ORBIT IN CARGO BAY
0.230	-13.672	-5.8146	29.497	5.2539	5.2544	20.905	5.0601	HALF ORBIT IN CARGO BAY
0.306	-11.763	-5.4523	27.079	6.1855	6.1859	23.742	4.8645	HALF ORBIT IN CARGO BAY
0.383	-10.175	-10.847	20.448	2.6768	2.6771	18.386	1.1925	HALF ORBIT IN CARGO BAY
0.459	-9.2107	-18.611	12.087	-3.0253	-3.0251	9.6517	-4.2705	HALF ORBIT IN CARGO BAY
0.536	-8.8396	-25.749	4.4366	-8.6100	-8.6099	1.7170	-9.6599	HALF ORBIT IN CARGO BAY
0.612	-8.9669	-32.358	-2.6213	-14.044	-14.044	-5.5577	-14.933	HALF ORBIT IN CARGO BAY
0.689	-9.5119	-38.512	-9.1772	-19.308	-19.308	-12.280	-20.066	HALF ORBIT IN CARGO BAY
0.766	-10.406	-44.270	-15.302	-24.395	-24.395	-18.531	-25.044	HALF ORBIT IN CARGO BAY
0.766	-10.406	-44.270	-15.302	-24.395	-24.395	-18.531	-25.044	HALF ORBIT AFTER DEPLOYED
0.842	-11.595	-50.559	-16.653	-30.601	-30.601	-26.202	-30.107	HALF ORBIT AFTER DEPLOYED
0.919	-13.026	-56.211	-19.130	-36.222	-36.222	-34.186	-32.999	HALF ORBIT AFTER DEPLOYED
0.995	-14.649	-61.201	-22.618	-41.340	-41.340	-42.342	-34.098	HALF ORBIT AFTER DEPLOYED
1.072	-16.417	-65.009	-26.193	-46.022	-46.022	-50.006	-34.006	HALF ORBIT AFTER DEPLOYED
1.148	-18.189	-67.217	-16.462	-50.161	-50.161	-56.863	-33.202	HALF ORBIT AFTER DEPLOYED
1.225	-19.869	-67.037	-9.8337	-53.541	-53.541	-62.721	-32.282	HALF ORBIT AFTER DEPLOYED
1.301	-21.446	-64.170	-5.5785	-56.088	-56.088	-67.620	-31.546	HALF ORBIT AFTER DEPLOYED
1.378	-22.909	-59.066	-3.1292	-57.929	-57.929	-71.479	-31.235	HALF ORBIT AFTER DEPLOYED
1.454	-24.256	-52.701	-1.4446	-59.311	-59.311	-73.284	-32.873	HALF ORBIT AFTER DEPLOYED
1.531	-25.487	-45.382	-0.26810	-60.313	-60.313	-73.103	-36.385	HALF ORBIT AFTER DEPLOYED
1.531	-68.930	-75.176	-15.752	-89.454	-89.454	-79.897	-79.897	ORBITAL AVERAGE AFTER DEPLOYED

Conclusion: This problem verifies that this installation of the SINDA/FLUINT program performs exactly as the authors intended. Computer system output files are included in the applicable appendix to QMI.1000.001 (Q-Metrics 1999).

4C.4.2 PEER REVIEW OF SINDA/FLUINT MODEL AND INPUT DATA

The specific thermal model of the MCO and the transportation cask was developed by Q-Metrics, Incorporated, of Woodinville, Washington. Although Q-Metrics is on the Evaluated Suppliers List and the analysis work was conducted in accordance with its internal quality assurance procedures, an additional level of V&V was achieved by having the model and the model inputs and output independently reviewed. The peer review was provided by Waste Management Federal Services, Northwest Operation, as part of the safety analysis report for packaging for the MCO transportation cask (Smith 2000).

The peer review consisted of complete review of the input and output files, including the construction of the model, the selection of material thermal properties, the mathematical derivations used to calculate convection heat transfer coefficients as a function of local ambient conditions, and the reaction rate equations used to define the chemical corrosion heat generation and associated gas generation. A copy of the signed review checklist is provided in Figure 4C-20.

4C.5 GOTH_SNF PREDICTIONS VERSUS FIRST ARTICLE DRYING TEST RESULTS

The GOTH_SNF code was used to simulate the first article drying tests. Cold vacuum drying tests were conducted in 1998 and early 1999 using a 'first article' MCO, fuel and scrap baskets, and CVDF skid-mounted equipment. The spent fuel payload was simulated using carbon steel tubing for the fuel assemblies and crushed carbon steel pipe for the scrap. Water-holding cavities were formed at the tops of selected simulated fuel assemblies to mimic damaged fuel assemblies. The results of the testing and the post-test analytical simulations of the tests, presented in SNF-4083 (Crea and Fryer 1999), demonstrated that the CVDF equipment and processes would remove the free water remaining after MCO draining within the time allotted in the CVDF process plan.

Table 4C-8 (a reproduction of Table 3.3 from SNF-4083) summarizes the close match between the GOTH_SNF predicted drying times and the tested results for seven of the tests. The tested conditions included variations in the amount of residual water after draining from 12.0 L to 22.6 L, the distribution of the water within the MCO, simulation of continuous vacuum drying versus the 8-4-4 cycle (i.e., 8 hours of initial vacuum drying, followed by a continued cycle of 4 hours of helium purge at atmospheric pressure, and then 4 hours of vacuum drying until the MCO is dried), and variation in the MCO annulus water temperature. The tested MCO configuration consisted of one scrap basket in the top position and four fuel baskets.

Table 4C-8 presents the comparison between the model and test results for three time points during the drying cycle: the time for dryout of the simulated scrap basket, the time to reach 9 torr pressure, and the time to reach 1 torr pressure within the MCO. The table also presents the test time required to reach 0.1 torr pressure and the time to reach the level of dryness required for the 1-hour pressure rise criteria. These later two times were not simulated in GOTH_SNF. As seen from the Table 4C-8, the GOTH_SNF predicted drying times compare favorably over the seven tested conditions.

Table 4C-8. Comparison of Drying Times Between First Article Tests and GOTH_SNF Code Predictions.

Test #	Drying Cond	Oper Mode/Variations	Resid Water	Water Dist	Cycle Vac/hol/v ac -hrs	Test Dryin Time -last scrap temp drop	Model Calc Dryin Time -- align scrap =0.0	Test 9 torr .174 psia Time	Model 9 torr .174 psia Time	Test 1 torr .02 psia Time	Model 1 torr 0.2 psia Time	Test .1 torr .002 psia Time	Test time to pass first 1 hr press rise test
liter						hrs	hrs	hrs	hrs	hrs	hrs	hrs	hrs
fa14	Nom	Base	12.0	un/cav	8/4/4	15.0	14.2	14.3	14.2	15.2	14.3	28.3	34.0
fa20	Nom	Rev.1	12.0	un/cav	8/4/4								
fa17c	Nom	Wet flit	12.0	un/cav	8/4/4								
fa16a	Nom	cont pmp	12.0	un/cav	cnt vac	12.5	11.3	11.8	11.8	12.8	11.8	19.1	25.1
fa16b	Nom	cont pmp	12.0	un/cav	cnt vac								
fa15	Nom	hel inj	12.0	un/cav	8/4/4	13.6	Bad D	13.1	13.1	13.8	14.8	20.5	26.5
fa18	Nom	Tcond	12.0	un/cav	8/4/4								
fa19	Nom	cont hre	12.0	un/cav	cnt vac								
fa17a	Nom	Tbath	12.0	un/cav	8/4/4	28.3	23.0	23.4	23.3	28.3	23.4	36.5	42.5
fa17b	Nom	Tbath	12.0	un/cav	8/4/4								
fa21	Nom	cont pmp	12.0	un/cav	cnt vac	13.9	12.6	13.1	13.1	14.3	13.1	29.2	35.2
fa6	Good	Base	7.0	uniform	8/4/4	---	2.9	3.4	3.4	3.5	3.4	22.9	28.7
fa11	Poor	Base	22.6	cav	8/4/4	28.6	28.1	28.3	28.3	28.7	29.7	45.5	50.8

Array: dc1

Further evidence of the favorable match between the GOTH_SNF predictions and the test data is illustrated in Figures 4C-21 and 4C-22. These figures present a comparison of predicted transient temperatures versus test data for the upper scrap basket and a middle fuel basket, respectively, for an 8-4-4 vacuum drying cycle. The effects of the evaporative cooling under this varying MCO pressure cycle can especially be seen in the temperature trends for the scrap basket (i.e., Figure 4C-21). Here the scrap basket temperature is seen to drop from the post-draining temperature of 120 °F to approximately 80 °F during the first 8 hours of vacuum drying. The temperature rebounds to approximately 110 °F under the 4-hour helium purge cycle as evaporative cooling is reduced by the elevated pressure and the thermal conductivity is improved between the scrap and the MCO wall because of the presence of the helium gas. The temperature cycle is repeated under the subsequent vacuum-helium purge cycles with the minimum temperature seen in each cycle decreasing as the water content is reduced. Overall, the good temperature match between the predicted and test data demonstrates that the GOTH_SNF model accurately simulates the heat transfer mechanisms within the MCO, the thermal mass of the system, and the evaporation process. For a full discussion of the comparison between the model and test results, see SNF-4083 (Crea and Fryer 1999).

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Figure 4C-1. Oxidation of Uranium and K Basin Fuel In Moist-Inert Gas.

Figure 4C-2. Level of Conservatism Produced by the Combination of Enhancement Factors and Literature-Based Reaction Rates.

Figure 4C-3. Constitutive Relation Validation.

Table 7-1: Constitutive Relation Validation				
A. Steam Table State [Keenan, et al, 1978]				
v (m³/kg)	P (MPa)	T (°C)	u (kJ/kg)	Δu (kJ/kg)
0.4011	0.46	145	2551.8	0
0.4010	0.72	360	2895.6	343.8
0.4011	1.00	600	3296.8	745.0
0.4034	1.80	1300	4679.5	2127.7
B. Calculated State				
$\Delta U = \Delta u/v$ (kJ)	P (MPa)	T (°C)		
0.0	0.46	145		
857.4	0.723	364		
1857.9	1.005	606		
5306.0	1.819	1309		

This table reproduced from Plys, M. G., S. J. Malinovic, M. Epstein, and D. R. Duncan, 1999, *HANSF 1.3 User's Manual*, SNF-3650, Rev. 1, Fluor Daniel Hanford, Incorporated, Richland, Washington.

Figure 4C-4. Fog Formation Sample Problem.

Table 7-2: Fog Formation Sample Problem				
A. Initial Quantities				
Temperature	T_o	323	K	
Saturation Pressure	P_{sat}	12350	Pa	
Pressure Derivative	dP_{sat} / dT	636	Pa/K	
Internal Energy Change	u_{fg}	2.23	MJ/kg	
Steam Mole Fraction		0.15		
Nitrogen Mole Fraction		0.85		
Steam Specific Heat		1409.	J/kgK	
Nitrogen Specific Heat		739.	J/kgK	
Steam Mass	m_{to}	0.1006	kg	
Relative Humidity	RH_{to}	1.2		
Nitrogen Mass		0.8871	kg	
Gas Heat Capacity	C_o	798	J/K	
B. Analytical Solution				
Parameter	θ	4.840E-2		
Temperature Rise		3.80	K	
Aerosol Mass		1.36E-3	kg	
C. Code Output				
Temperature Rise		3.82	K	
Aerosol Mass		1.373E-3	kg	

This table is reproduced from Plys, M. G., S. J. Malinovic, M. Epstein, and D. R. Duncan, 1999, *HANSF 1.3 User's Manual*, SNF-3650, Rev. 1, Fluor Daniel Hanford, Incorporated, Richland, Washington.

Figure 4C-5. Comparison of the Semi-infinite Slab Solution for Surface Temperature with Model Results.

Table 7-4: Comparison of the Semi-Infinite Slab Solution for Surface Temperature (T) with Model Results		
Time (seconds)	Model (°C)	Analytical (°C)
0	200.000	200.000
1000	172.715	170.75
2000	164.795	163.05
3000	159.645	158.25
4000	155.817	154.75
5000	152.779	151.95
10000	143.220	143.45
20000	134.046	134.25
30000	129.125	127.65
40000	125.902	122.55
50000	123.572	118.35

This table is reproduced from Plys, M. G., S. J. Malinovic, M. Epstein, and D. R. Duncan, 1999, *HANSF 1.3 User's Manual*, SNF-3650, Rev. 1, Fluor Daniel Hanford, Incorporated, Richland, Washington.

Figure 4C-6. Uranium Oxidation Validation.

Table 7-6: Uranium Oxidation Validation					
Case No.	Correlation (mg/cm ² -hr)	Calculated Mass Consumption (kg) ⁽¹⁾	HANSF Mass Consumption (kg)	Calculated ΔT (°C)	HANSF ΔT (°C) ⁽³⁾
1	Ritchie $10^{\left(13.8808 - \frac{5.7696 \times 10^3}{T}\right)}$	7.23×10^{-4} (2)	7.2×10^{-4} (2)	0.018	0.017 - 0.018
2	Pearce $10^{\left(6.1931 - \frac{2963}{T} + 0.3 \log\left(\frac{P_{H_2}}{1000}\right)\right)}$	0.0143 (4)	0.0144 (4)	0.36	-0.38
3	Pearce $10^{\left(4.33 - \frac{2144}{T} + 0.5 \log\left(\frac{P_{H_2}}{1000}\right)\right)}$	0.0423 (2)	0.0403 (2)	0.51	0.47 - 0.50
4	Pearce (Same as in Case 3)	0.0142 (4)	0.0142 (4)	0.17	0.15

This table is reproduced from Plys, M. G., S. J. Malinovic, M. Epstein, and D. R. Duncan, 1999, *HANSF 1.3 User's Manual*, SNF-3650, Rev. 1, Fluor Daniel Hanford, Incorporated, Richland, Washington.

Figure 4C-7. Comparison of Water Vapor and Total Pressures Between HANSF and GOTH_SNF, Normal Vacuum Drying Conditions.

Figure 4C-8. HANSF Results for Normal Vacuum Drying Conditions,
30 ft³/min Vacuum plus 0.7 ft³/min Helium Purge. (2 sheets)

VPXP7: VAC=30 cfm, He=0.7 scfm, U=10 UH3=12 H2O=26.5 Kg, ILAW=3

Figure 4C-8. HANSF Results for Normal Vacuum Drying Conditions, 30 ft³/min Vacuum plus 0.7 ft³/min Helium Purge. (2 sheets)

VPXP7: VAC=30 cfm, He=0.7 scfm, U=10 UH3=12 H2O=26.5 Kg, ILAW=3

Figure 4C-9. GOTH_SNF Results for Normal Vacuum Drying Conditions,
30 ft³/min Vacuum plus 0.7 ft³/min Helium Purge. (2 sheets)

Full Vacuum With Helium Purge of 0.7 acfm
Mon Sep 20 10:52:38 1999
GOTH Version 5.0.

Full Vacuum With Helium Purge of 0.7 acfm
Mon Sep 20 10:53:30 1999
GOTH Version 5.0.

Figure 4C-9. GOTH_SNF Results for Normal Vacuum Drying Conditions,
30 ft³/min Vacuum plus 0.7 ft³/min Helium Purge. (2 sheets)

Full Vacuum With Helium Purge of 0.7 scfm
Mon Sep 20 10:52:55 1999
GOTH Version 5.0

Figure 4C-10. Comparison of Water Vapor Pressure Between HANSF and GOTH_SNF, Degraded Vacuum Drying Conditions.

Figure 4C-11. HANSF Results for Degraded Vacuum Drying Conditions, 30 ft³/min Vacuum with No Helium Purge. (2 sheets)

VX24X: VAC=30 cfm, He=0.0 scfm, Netw U=10 UH3=12 H2O=26.5 Kg, ILAW=3

Figure 4C-11. HANSF Results for Degraded Vacuum Drying Conditions, 30 ft³/min Vacuum with No Helium Purge. (2 sheets)

Figure 4C-12. GOTH_SNF Results for Degraded Vacuum Drying Conditions,
30 ft³/min Vacuum With No Helium Purge. (2 sheets)

Pure Vacuum Pumping, Uniform Water Distribution, Bounding Conditions
Mon Dec 14 09:24:12 1998
GOTH Version 4.0 - June 1998

Pure Vacuum Pumping, Uniform Water Distribution, Bounding Conditions
Mon Dec 14 09:25:18 1998
GOTH Version 4.0 - June 1998

Figure 4C-12. GOTH_SNF Results for Degraded Vacuum Drying Conditions,
30 ft³/min Vacuum With No Helium Purge. (2 sheets)

Pure Vacuum Pumping, Uniform Water Distribution, Bounding Conditions
Mon Dec 14 09:24:38 1998
GOTH Version 4.0 - June 1998

Figure 4C-13. Schematic Diagram of a Semi-infinite Slab Subject to Convection.

Figure 4C-14. Normalized Temperature Distributions for a Semi-Infinite Slab Subject to Convection.

Figure 4C-15. Schematic Diagram for a Composite Cylinder with Uniform Heat Generation (Inner Cylinder) and Convection.

Figure 4C-16. Schematic Diagram of a Spent Nuclear Fuel Cask Subject to Radiant Heat Transfer Conditions.

Figure 4C-17. Schematic Diagram of Node Designations for Verification Problem Four.

Figure 4C-18. Benchmark Transient Thermal Performance for Problem Four.

Figure 4C-19. Transient Thermal Results Provided with the Sample Problem Used as Verification Problem Four.

Sample Problem Results

SUMMARY TABLE OF SINDA SAMPLE PROBLEM: SIMPLIFIED SATELLITE DEPLOYMENT ** TEMPERATURES **

TIME	CUBE	OUTSIDE	INSIDE	OTHER SIDE PANELS-----				STATUS SUMMARY
0.000	-17.232	-34.666	-3.3727	-19.258	-19.257	-14.281	-14.282	ORBITAL AVERAGE IN CARGO BAY, E= 0.80
0.000	-17.232	-34.666	-3.3727	-19.258	-19.257	-14.281	-14.282	HALF ORBIT IN CARGO BAY, E= 0.80
0.077	-16.744	-21.775	16.740	-8.2463	-8.2456	-2.3930	-4.5260	HALF ORBIT IN CARGO BAY, E= 0.80
0.133	-15.457	-12.044	27.115	1.20239E-02	1.26038E-02	9.9439	1.8986	HALF ORBIT IN CARGO BAY, E= 0.80
0.230	-13.667	-5.0120	29.499	5.2557	5.2562	20.908	5.0623	HALF ORBIT IN CARGO BAY, E= 0.80
0.306	-11.758	-5.4500	27.081	6.1873	6.1876	23.744	4.8665	HALF ORBIT IN CARGO BAY, E= 0.80
0.393	-10.170	-10.845	20.450	2.6788	2.6790	18.388	1.1946	HALF ORBIT IN CARGO BAY, E= 0.80
0.459	-9.2962	-18.609	12.089	-3.0234	-3.0232	9.6535	-4.2607	HALF ORBIT IN CARGO BAY, E= 0.80
0.536	-8.0353	-25.748	4.4379	-8.6082	-8.6081	1.7187	-9.6582	HALF ORBIT IN CARGO BAY, E= 0.80
0.612	-6.9629	-32.357	-2.6199	-14.042	-14.042	-5.5560	-14.932	HALF ORBIT IN CARGO BAY, E= 0.80
0.689	-9.5081	-38.511	-9.1759	-19.306	-19.306	-12.278	-20.064	HALF ORBIT IN CARGO BAY, E= 0.80
0.766	-10.402	-44.269	-15.301	-24.394	-24.394	-18.530	-25.042	HALF ORBIT IN CARGO BAY, E= 0.80
0.766	-10.402	-44.269	-15.301	-24.394	-24.394	-18.530	-25.042	HALF ORBIT AFTER DEPLOYED, E= 0.80
0.842	-11.591	-50.557	-16.652	-30.600	-30.600	-26.200	-30.105	HALF ORBIT AFTER DEPLOYED, E= 0.80
0.919	-13.023	-56.210	-19.129	-36.221	-36.221	-34.185	-32.998	HALF ORBIT AFTER DEPLOYED, E= 0.80
0.995	-14.645	-61.200	-22.617	-41.339	-41.339	-42.341	-34.097	HALF ORBIT AFTER DEPLOYED, E= 0.80
1.072	-16.414	-65.008	-26.192	-46.021	-46.021	-50.005	-34.005	HALF ORBIT AFTER DEPLOYED, E= 0.80
1.148	-18.186	-67.216	-16.462	-50.160	-50.160	-56.862	-33.201	HALF ORBIT AFTER DEPLOYED, E= 0.80
1.225	-19.866	-67.036	-9.8330	-53.540	-53.540	-62.720	-32.281	HALF ORBIT AFTER DEPLOYED, E= 0.80
1.301	-21.443	-64.170	-5.5779	-56.087	-56.087	-67.619	-31.546	HALF ORBIT AFTER DEPLOYED, E= 0.80
1.378	-22.907	-59.044	-3.1287	-57.928	-57.928	-71.478	-31.235	HALF ORBIT AFTER DEPLOYED, E= 0.80
1.454	-24.253	-52.701	-1.4441	-59.311	-59.311	-73.283	-32.872	HALF ORBIT AFTER DEPLOYED, E= 0.80
1.531	-25.484	-45.381	-0.24758	-60.312	-60.312	-73.102	-36.384	HALF ORBIT AFTER DEPLOYED, E= 0.80
1.531	-68.930	-75.176	-15.752	-89.454	-89.454	-79.897	-79.897	ORBITAL AVERAGE AFTER DEPLOYED, E= 0.80

Figure 4C-20. SINDA/FLUINT Model Checklist Signoff for Peer Review
Conducted as Part of the Safety Analysis Report for
Packaging of the Multi-Canister Overpack.

CHECKLIST FOR CHECKING OF ANALYSIS/CALCULATIONS			
Document Number/Revision: <u>HNF-SD-TP-SARP-017, Rev. 1</u>			
Document Title: <u>Thermal Evaluation (Chapter 8-8)</u>			
Yes	No	N/A	
☒	☐	☐	Problem completely defined.
☒	☐	☐	Appropriate analytical method used.
☒	☐	☐	Necessary assumptions are appropriate and explicitly stated.
☒	☐	☐	Computer codes and data files documented.
☒	☐	☐	Data used in calculations explicitly stated in document.
☒	☐	☐	Sources of non-standard formula/data are referenced and the correctness of the reference verified.
☒	☐	☐	Data checked for consistency with original source information as applicable.
☒	☐	☐	Mathematical derivations checked including dimensional consistency of results.
☒	☐	☐	Models appropriate and used within range of validity or use outside range of established validity justified.
☐	☐	☒	Hand calculations checked for errors.
☒	☐	☐	Code run streams correct and consistent with analysis documentation.
☒	☐	☐	Code output consistent with input and with results reported in analysis documentation.
☒	☐	☐	Acceptability limits on analytical results applicable and supported. Limits checked against sources.
☒	☐	☐	Safety Margins consistent with good engineering practices.
☒	☐	☐	Conclusions consistent with analytical results and applicable limits.
☒	☐	☐	Results and conclusions address all points required in the problem statement.
I have checked the analysis/calculation and it is complete and accurate to the best of my knowledge.			
Engineer/Checker <u><i>[Signature]</i></u>			Date <u>5/5/98</u>
NOTE: Any hand calculations, notes or summaries generated as part of this check should be signed, dated, and attached to this checklist. Material should be labeled and recorded so that it is intelligible to a technically qualified third party.			

Figure 4C-21. GOTH_SNF Predicted versus First Article Test Data for Scrap Basket.

Figure 4C-22. GOTH_SNF Predicted versus First Article Test Data for Middle Fuel Basket.

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ATTACHMENT 4C-A
QUALIFICATIONS OF PEER REVIEWERS

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ATTACHMENT 4C-A
QUALIFICATIONS OF PEER REVIEWERS

The following are abridged versions of the resumes for the independent reviewers. In some cases, the list of publications and presentations has been shortened for the sake of brevity.

**CURRICULUM VITAE
GEORGE APOSTOLAKIS**

POSITION

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EDUCATION

Diploma, Electrical Engineering, National Technical University, Athens, Greece, June 1969

M.S., Engineering Science, California Institute of Technology, Pasadena, California, June 1970

Ph.D., Engineering Science and Applied Mathematics, California Institute of Technology, Pasadena, California, June 1973

HONORS AND SPECIAL RECOGNITION RECEIVED

Mark Mills Award, American Nuclear Society, 1974

Effective Teaching Award, School of Engineering and Applied Science, UCLA, 1977

IBM Visiting Distinguished Scholar, Northeastern University, 1985

Opening Lecture at the 1987 Symposium of the Scandinavian Society of Reliability Engineers, Denmark, 1987

Certificate of Appreciation, U.S. Nuclear Regulatory Commission, 1990

Outstanding Service Award, Society for Risk Analysis, 1991

Fellow, American Nuclear Society, 1992

Fellow, Society for Risk Analysis, 1992

Honorary Chairman, PSA '93, Clearwater, FL, 1993

Invited Luncheon Speaker, "Observations on the Nuclear Regulatory Commission's Risk-Informed Regulatory Initiative," Society for Risk Analysis Annual Meeting, Washington, DC., December 7-10, 1997.

RESEARCH INTERESTS

Methods for risk and reliability assessment of complex technological systems; uncertainty analysis; decision analysis; risk management involving several stakeholder groups; waste management; fire risk assessment; human reliability models; organizational factors and safety culture; software dependability.

CONSULTING ACTIVITIES

U.S. Nuclear Regulatory Commission (1977-1979)

Member of Research Review Group, Probabilistic Analysis Branch.

Lawrence Livermore National Laboratory (1978-1982)

Seismic Safety Margins Research Program.

Sandia National Laboratories (1979-1980)

Risk Methodology for Geologic Disposal of Radioactive Waste.

Bechtel Power Corporation (1980-1985)

Probabilistic Risk Analysis.

USNRC/Lawrence Livermore National Laboratory (1980-1982)

Member, Peer Review Panel, Load Combination Program.

Pickard, Lowe and Garrick, Inc. (1977-1987)

Probabilistic Risk Analysis of the Oyster Creek, Zion, Indian Point, Browns Ferry, Midland and Seabrook Nuclear Generating Stations; Principal Responsibilities: methodology, data analysis, human error modeling, and fire risk assessment; Member of Technical Review Board, Seabrook Probabilistic Safety Study; Review of the Use of Expert Opinions in PRA.

State of California; Office of the Attorney General (representing Governor G. Deukmejian) (1983)

Expert Witness before the Atomic Safety and Licensing Appeal Board, Diablo Canyon Nuclear Power Plant; Testimony: On the Use of Probability Theory and Statistics in a Design Verification Program.

Lawrence Livermore National Laboratory (1983-1984)

Review of the Fire Risk Analysis of the Millstone 3 PRA.

Rockwell International (1983-1984)

Probabilistic Risk Analysis; Common Cause Failures; Quantitative Safety Goals.

International Atomic Energy Agency (1984-1987)

Manual for Probabilistic Safety Assessment (PSA) Utilization and Implementation for Safety Decisions.

Commission of the European Communities, Ispra Joint Research Center (1986)

Benchmark Exercises on Common Cause Failures, Human Errors, and Industrial Risk.

Swedish Nuclear Power Inspectorate (1986-1987)

Common Cause Failures, Fire Risk Assessment, Data Analysis.

Jet Propulsion Laboratory (1986-1987)
SP-100 Project, Reliability, Availability, Maintainability.

State of California: Department of Health Services (1987)
Member, Scientific Advisory Committee, IT Panoche Hazardous Waste Facility.

Factory Mutual Research Corporation (1987-Present)
Probabilistic Risk Analysis.

U.S. Nuclear Regulatory Commission (1987-1988)
Member, NUREG-1150 Peer Review Committee.

The Ralph M. Parsons Company (1988-1990)
Chemical Stockpile Disposal Program, Fire Risk Analysis.

Electric Power Research Institute (1988-1990)
Fire Risk Analysis.

U.S. Department of Energy, Advisory Committee on Nuclear Facilities Safety, (1988-1990)
Probabilistic Risk Assessment.

Sandia National Laboratories (1988-1990)
Probabilities of Events and Processes Affecting Geologic Repositories, Model
Uncertainties, Expert Opinions.

State of Minnesota, Waste Management Board (1988-1990)
Review of Reliability Study of a Proposed Hazardous Waste Stabilization and Containment
Facility.

U.S. Department of Energy, Office of the Director, New Production Reactors, Senior Advisor
(1989-1992)
Program Management, Probabilistic Risk Assessment.

Battelle Pacific Northwest Laboratories, Advanced LWR Project Steering Committee, Member
(1990-1991)
Advanced Power Reactors.

Sandia National Laboratories (1991-1994)
PRA Working Group

DOE/NRC/EPRI (1992-1995)
Member, Senior Seismic Hazard Analysis Committee

MITRE Corporation (1994-Present)
Expert Panel on Risk Assessment, Tooele Chemical Demilitarization Facility

PUBLIC SERVICE

National Research Council (1986-1988)

- Committee to Assess the Safety and Technical Issues at DOE Reactors.
- U.S. Nuclear Regulatory Commission (1989-1990)
Peer Review Panel of NUREG-1150 Final Report.
- National Research Council (1992-1993)
Committee on Alternative Chemical Demilitarization Technologies
- State of California, Water Resources Control Board (1995-1996)
SB 1764 Advisory Committee
- International Association for Probabilistic Safety Assessment and Management (1994-Present)
Founder and Secretary
- U.S. Nuclear Regulatory Commission (1995-Present)
Advisory Committee on Reactor Safeguards

NED E. BIBLER

Senior Advisory Scientist
Savannah River Technology Center, Bldg. 773-A
Westinghouse Savannah River Company
Aiken, SC 29808

Phone: (803)725-2313 Fax: (803)725-4704 E-mail: ned.bibler@srs.gov

EDUCATION

Ph.D. Radiation Chemistry, Ohio State University, Columbus, Ohio, 1965
M.S. Analytical Chemistry, Ohio State University, Columbus, Ohio, 1963
B.S. Chemistry, Denison University, Granville, Ohio, 1959

WORK HISTORY

1965 - present:

Westinghouse Savannah River Company, Savannah River Technology Center & E. I. Du Pont de Nemours & Co., Savannah River Laboratory

Primary research and development directed to radioactive glass & waste characterization, radioactive Defense Waste Processing Facility process demonstrations, effects of radiation on waste forms, and processes and support of separation processes.

1962 - 1965:

DOE Graduate Research Scientist, Ohio State University
Tritium beta ray induced decomposition of heavy water vapor.

INTERESTS & EXPERIENCE

Authority in radiation chemistry - effects of radiation on materials and processes. Managed team that prepared kilogram amounts of glass containing actual radioactive SRS waste using DWPF process. Developed and perfected DWPF durability test to be performed remotely on radioactive glass in the shielded cells. Demonstrated method to measure all 24 radionuclides in DWPF glass for repository acceptance. Measured effects of alpha, beta, gamma, alpha recoil, and heavy ion radiations on DWPF glass. Initiated ERDA contract with Ga. Tech. and MPO with ANL for irradiation of hundreds of gallons of TPB. Measured effect of radiation on glass leaching. Measured radiolytic H₂ production from materials such as concrete, organic resins, and various solutions. The only scientist to develop method for catalyzed decomposition of oxalate to decrease the amount of organic oxalate going to SRS waste tanks; to demonstrate radiolytic instability of Fe(II) ions in HEM solvent extraction solutions. (This significantly improved the process to separate Np and Pu.); to measure radiolytic decomposition of PuF₆, a volatile Pu compound; to measure radiolytic decomposition of water by fission fragments from Cf-252. Demonstrated the use of nuclear weapons as pulse radiolysis sources. Measured radiation dose rate effect on fast organic radical reactions in liquid halocarbons.

Elucidated the mechanism and measured the radiolytic rate of flammable H₂ formation in SRS HLW storage and process solutions. These rates are currently used in safety calculations at SRS in the DWPF, the tank farm, and F and B line storage and process vessels.

Was one of the first scientists in ~ 1975 to emphasize to the scientific community the hazards associated with radiolytic hydrogen production during storage of nuclear wastes containing hydrogen compounds (water or organics). This led to TRU waste drums being vented during storage and was one of the factors for glass being favored as the immobilization matrix for high level wastes.

Ned E. Bibler

PROFESSIONAL SOCIETY ACTIVITY

- ASTM committee C26 on Nuclear Fuel Cycle (1993-present) - chair subcommittee C26.13 on standards for geologic storage of nuclear waste. Instrumental in getting 2 standards accepted - C 1285-94, "Standard Test Methods for Determining Chemical Durability of Nuclear Waste Glasses: The Product Consistency Test, and C 1174-91, "Standard Practice for the Long-Term Behavior of Waste Package Materials Including Waste Forms Used in the Geologic Disposal of High-Level Waste". (1987 to present)
- Member of University of Notre Dame's Radiation Chemistry Data Center Advisory Board; (1983-1990) Chairman (1989-1990)
- American Chemical Society (1965-Present) - Chairman of Savannah River Section. (~1975)
- Member of Organizing Committee for Materials Research Society Symposium on Scientific Basis for Nuclear Waste Management (1986-1994)
- Member of Executive Organizing Committee for Materials Research Society Symposium on Scientific Basis for Nuclear Waste Management (1994-Present)
- Member of Organizing Committee for International Waste Management Symposia in Tucson, AZ. (1987-Present)
- Member of USDOE Science Advisory Panel on behavior of Hanford HLW in storage tanks. (1991-1994) Member of USNRC Panel consulting on fate of Cs and I in reactor loss of coolant accident. (1985-1987)

AWARDS & HONORS

- Selected to participate in the DOE Office of Basic Energy Sciences Workshop on "Research Needs and Opportunities in Radiation Chemistry." Scheduled for April 19, 1998
- Invited speaker to Westheimer Peace Symposium at Wilmington College in Ohio. Subject "High Level Nuclear Waste - A Positive Way to Deal with It". (1995)
- Chosen to be member of third American delegation to visit Chelyabinsk-65 in Siberia, Russia. This is one of Stalin's secret sites for Pu production. Purpose was to establish future technical exchanges.
- George Westinghouse Bronze Signature Award of Excellence (Applying ICP-MS to analyzing high level wastes.) (1991)
- George Westinghouse Bronze Signature Award of Excellence (DWPF durability test, PCT) (1989)
- Invited speaker at the First Hanford Separation Science Workshop; subject "Radiation Effects on Separations Materials and Processes" (1991)
- Invited participant in Gordon Research Conferences on Radiation Chemistry (1971-1994) ¹⁹⁹³

SELECTED PUBLICATIONS**Peer Reviewed Papers:**

1. "Determination of Long-Lived Fission Products and Actinides in Savannah River Site HLW Sludge and Glass for Waste Acceptance", N. E. Bibler, W. F. Kinard, W. T. Boyce, and C. J. Coleman, accepted for publication in Journal of Radioanalytical and Nuclear Chemistry (1998).
2. "Radiochemical Analyses For The Defense Waste Processing Facility Startup At The Savannah River Site." W. F. Kinard, N. E. Bibler, C. J. Coleman, and R. A. Dewberry, Journal of Radioanalytical and Nuclear chemistry, Vol. 219, No. 2, pp. 197-201, (1997).
3. "Characterization Of The Radioactive Glass Currently Being Produced By The DWPF At Savannah River Site", N. E. Bibler, J. W. Ray, T. R. Fellingner, O. B. Hodoh, R. S. Beck, and O. G. Lien, Waste Management '98, March 1998.
4. "Durabilities and Initial Radiation Damage Studies of Glasses for Immobilization of Plutonium", N. E. Bibler, J. C. Marra, W. G. Ramsey, J. M. Pareizs, T. F. Meaker and M. H. Tosten, accepted for presentation and publication at, Waste Management '96, February 1996.

Ned E. Bibler

5. "Inductively Coupled Plasma-Mass Spectrometry Studies of the Chemistry of Fission Products and Actinides in High Level Wastes: Lessons That Can Be Applied to Environmental Measurements", W. F. Kinard, N. E. Bibler, C. J. Coleman, and S. J. Wyrick, *Radiochimica Acta* 66/67, pp. 259-263, (1994). This paper illustrates that the relative concentrations of U-235 fission products in SRS and Hanford wastes can be explained by their fission yields, solubilities in caustic, and differences in HLW waste management methods at the two sites.
6. "Development of an ASTM Standard Glass Durability Test, The Product Consistency Test (PCT), for High-Level Radio-active Waste Glass", C. M. Jantzen, N. E. Bibler, D. C. Beam, W. G. Ramsey, *Nuclear and Hazardous Waste Management, Spectrum '94*, American Nuclear Society, pp. 164-169, (1994).
7. "Determination of Noble Metals in Savannah River Site High-Level Radioactive Sludge, C. J. Coleman, W. F. Kinard, N. E. Bibler, D. F. Bickford, and W. G. Ramsey, *Waste Management '93*, pp. 767-771, (1993). This was the first demonstration of the determination of noble metal fission products in HLW.
8. "Radiation Effects on Separations Materials and Processes", N. E. Bibler, Document No. PNL-SA-21775, *Invited Paper at Proceedings of the First Hanford Separation Science Workshop*, Richland, WA, pp. 11-23-27, (1993).
9. "Hydrogen Generation Rates In Savannah River Site High-Level Nuclear Waste", D. T. Hobbs, P. W. Norris, S. A. Pucko, N. E. Bibler, D. D. Walker, and P. d'Entremont, *Waste Management '92*, pp. 1063-1066, (1992). This was the first paper to quantify radiolytic H₂ production rates in HLW and to illustrate the effect of NO₃⁻ in the tanks.
10. "Recent Results on The Effect of Gamma Radiation on the Durability and Micro Structure of DWPF Glass, N. E. Bibler, M. H. Tosten, and D. C. Beam, *Proceedings of High-Level Radioactive Waste Management*, Vol. 2, pp. 1103-1109, American Nuclear Society, (1990).

Selected Presentations:

1. "Durabilities and Microstructures of Actual Radioactive Glasses to Immobilize Excess Actinides and Reprocessing Wastes at Savannah River Site", N. E. Bibler, W. G. Ramsey, T. F. Meaker, and J. M. Pareiza, accepted for presentation at the Material Research Society Symposium on Scientific Basis for Nuclear Waste Management, November, 1995.
2. "Applications of Inductively Coupled Plasma-Mass Spectrometry to Radiochemical Problems at the Savannah River Site, W. F. Kinard, N. E. Bibler, C. J. Coleman, W. T. Boyce, and S. B. Wyrick, presented at the DOE/ASTM Symposium, Gatlinburg, TN, (1994).
3. "The Product Consistency Test for the DWPF Waste Form", C. M. Jantzen and N. E. Bibler, *Westinghouse E. I. du Pont de Nemours & Co., Document No. WSRC-MS-90-149*, presented at the *Radioactive Waste Products Seminar*, Jülich, Federal Republic of Germany, (1990).

Assistant Professor Ken Czerwinski
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Massachusetts Institute of Technology
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Cambridge, MA 02139
USA

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Telefax: (617) 258-8863
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Education History

Ph.D. in Nuclear Chemistry
University of California, Berkeley
Berkeley, CA
August 1987 - June 1992

Citizenship: USA

B.A. in Chemistry
B.A. in Russian Language
Knox College
Galesburg, IL
August 1983 - June 1987

Employment History:

Assistant Professor
Department of Nuclear Engineering
Massachusetts Institute of Technology
From 13 November 1996

Associate Research Scientist
July 1992- September 1996
Institut für Radiochemie
Technische Universität München
Supervisor: Dr. Professor J.I. Kim

Postdoctoral Fellow
April 1992- July 1992
Glenn T. Seaborg Institute for Transactinium
Science
Lawrence Livermore National Laboratory
Supervisor: Professor Darleane C. Hoffman

Graduate Student Research Assistant
October 1987- March 1992
College of Chemistry
University of California, Berkeley
Berkeley, CA
Supervisors: Professor Darleane C. Hoffman,
Professor Kenneth N. Raymond

Research Assistant
August 1986-December 1986
June 1987-August 1987
Argonne National Laboratory
Argonne, IL
Supervisor: Dr. Robert Atcher

Additional Activities and Honors

Research Associate, Glenn T. Seaborg Institute for Transactinium Science 1997-1998.

Norman C. Rasmussen Career Development Chair, 1997.

Coordinator:
CEC-Humic Acid Program. Effect of Humic Substances on the Migration of Radionuclides:
Complexation of Actinides with Humic Substances
April 1994-May 1996
Duties include coordination and exchange of information amongst five European Union laboratories,
preparation of reports and proposals, and presentation of project results at conferences.

Visiting Scientist:

Forschungszentrum Rossendorf, Institut für Radiochemie

July 1994; July 1995, July 1996

Facilitated exchange of information and ideas involving environmental chemistry of actinides

Participant:

National Academy of Science Young Investigators Program on Nuclear Accidents and Radioactive Contamination

June 1993, June 1994

Program involved development of collaborative research program on radioactive contamination stemming from the Chernobyl accident with scientists from Belarus, Ukraine and the United States.

Experience:

Actinide spectroscopy

Actinide thermodynamics

Environmental chemistry of the actinide elements

Geochemical modeling

Development of actinide-transactinide separations

Low-level radiotracer handling and detection

Handling, preparation and chemistry of cyclotron irradiated materials

Handling and separation of actinides and fission products

Preparation and use of radiotracer solutions

Use of alpha, beta, and gamma radiation detectors

Synthesis and characterization of polydentate organic ligands

Synthesis of inorganic complexation agents

Development of radiotracer generators

Supervision of laboratory workers

Coordination of research programs

Research History:

The research program at MIT is centered on understanding the chemistry and behavior of actinides and fission products on the molecular level. The results are thermodynamically and kinetically interpreted to facilitate speciation and transport modeling. Determining the behavior and physical properties of relevant radionuclide species is performed to evaluate potential applications. Industrial and environmental systems containing radionuclides are examined. By understanding such systems, one can determine relevant species, study their behavior, verify laboratory results, test models, help focus efforts for further investigations, and incorporate the latest concepts into education. The research is applied to a number of fields concerning radioactive materials: radiation safety, dosimetry, remediation, disposition, and geochemical modeling.

Post-doctoral research was centered on the thermodynamic description of the speciation and transport of actinide elements in the environment. The complexation of actinides with humic acid with regards to geochemical modeling was the main project. Investigations utilize both radiochemical and spectroscopic techniques. Work was also performed on actinide complexation with a range of natural organic and inorganic ligands. Spectroscopy was employed to examine actinides in the aqueous and solid phase.

As a graduate student, was a joint member of two research groups; inorganic/synthetic chemistry and nuclear chemistry. Synthesis of polydentate ligands for tetravalent metal ions was conducted. Furthermore, uranyl extraction studies utilizing synthesized ligands were performed. Nuclear studies involved the examination of the fundamental chemistry of rutherfordium (element 104) and understand the relativistic effects on the chemistry of the transactinide elements. Work was also conducted on the synthesis of ^{263}Rf , a new isotope of Rf utilizing specific chemical separation procedure perfected while studying Rf chemistry with ^{261}Rf . Collaborative work with other members of the Lawrence Berkeley Laboratory nuclear science division was conducted.

Undergraduate work conducted at Argonne National Laboratory was in the nuclear medicine field. The studies included synthesis of alpha emitting bismuth based radiopharmaceuticals and the development of inorganic Bi-Pb generators. Chemical separations of cyclotron activated foils were also performed.

Publications:

J.F. McCarthy, K.R. Czerwinski, W.E. Sanford, P.M. Jardine, and J.D. Marsh: Mobilization of Actinides from Disposal Trenches by Natural Organic Matter. *J. Contam. Hydro.* In press.

K.R. Czerwinski: Environmental Behavior of Actinide Ions: Modeling the Influence of Humic Substances. *Proceed. Inter. Symp. Rad. Safety Manag.* '97. Eds. M.J. Song, M.C. Lee, and S.W. Shin. Taejon, Korea 1997 pg. 264.

M. Draye, A. Favre-Régouillon, J. Foos, A. Guy, and K.R. Czerwinski: Competitive Sorption of Lanthanum and Europium by Phenolic Resins Containing 8-Hydroxyquinoline. *Proceed. Inter. Symp. Rad. Safety Manag.* '97. Eds. M.J. Song, M.C. Lee, and S.W. Shin. Taejon, Korea 1997 pg. 423.

K.R. Czerwinski and J.I. Kim: Complexation of Transuranic Ions by Humic Substances: Application of Laboratory Results to The Natural System. *MRS Scientific Basis for Nuclear Waste Management XX*, Eds. W.J. Gray and I.R. Triay, 743 (1997).

I. Pashalidas, K.R. Czerwinski, and J.I. Kim: Liquid-Solid Phase Equilibria of Pu(VI) and U(VI) in Aqueous Carbonate Systems. Determination of the Carbonate Stability Constants. *Radiochim. Acta* 76, 55 (1997).

P. Zeh, K.R. Czerwinski, and J.I. Kim: Speciation of Uranium in Gorleben Groundwaters. *Radiochim. Acta* 76, 37 (1997).

K.R. Czerwinski, D.S. Rhee, G. Buckau, and J.I. Kim: Complexation of Trivalent Actinide Ions (Am^{3+} , Cm^{3+}) with Humic Acid. Influence of Ionic Strength. *Radiochim. Acta* 72, 179 (1996).

J.I. Kim and K.R. Czerwinski: Complexation of Metal Ions with Humic Acid: Charge Neutralization Model. *Radiochim. Acta* 73, 5 (1996).

McCarthy, J.F., Sanford, W.E., and Czerwinski, K.R.: Natural Organic Matter Transports Transuranic Radionuclides in Shallow Groundwater. *International Humic Substances Society Meeting*, Poland 1996

K. R. Czerwinski, G. Buckau, F. Scherbaum, and J.I. Kim: Complexation of the Uranyl Ion with Aquatic Humic Acid. *Radiochim. Acta* 65, 111 (1994).

K.R. Czerwinski, K. Gregorich, N.J. Hannink, C.D. Kacher, B.A. Kadkhodayan, S.A. Kreek, D. Lee, M. Nurmiä, A. Türler, G.T. Seaborg, and D.C. Hoffman: Solution Chemistry of Element 104. Part I. Liquid-Liquid Extractions with Triisooctylamine. *Radiochim. Acta* 64, 23 (1994).

K.R. Czerwinski, C.D. Kacher, K. Gregorich, T. Hamilton, N.J. Hannink, B.A. Kadkhodayan, S.A. Kreek, D. Lee, M. Nurmiä, A. Türler, G.T. Seaborg, and D.C. Hoffman: Solution Chemistry of Element 104: Part II. Liquid-Liquid Extractions with Tributyl Phosphate. *Radiochim. Acta* 64, 29 (1994).

M.M. Hindi, B. Sur, K.L. Wedding, D.W. Bardayan, K.R. Czerwinski, M.T.F. de Cruz, D.C. Hoffman, R.M. Larimer, K.T. Lesko, and E.B. Norman: β^+ Decay and Cosmic-Ray Half-Life of ^{91}Nb . *Phys. Rev. C* 47(6), 2598 (1993).

Zimmerman, H.P., Gober, M.K., Kratz, J.V., Schädel, M., Bruchle, W., Schimpf, E., Gregorich, K.E., Türler, A., Czerwinski, K.R., Hannink, N., Kadkhodayan, B., Lee, D.M., Nurmiä, M.J.,

Your Single Source for Chemical, Materials, and Environmental Technology

JAMES R. DIVINE

EXPERTISE

- Application of electrochemical and chemical engineering principles to industrial processes.
- Evaluation of the safe and proper use of engineering materials including the investigation of corrosion and degradation of metals and polymers in waste management, nuclear, construction, and industrial operations.
- Chemical behavior of high level wastes.
- Mitigation of materials degradation including cathodic protection and materials selection.
- Decontamination using chemical methods (chemical cleaning).
- Inter-disciplinary information exchange with emphasis on chemistry and engineering.

EDUCATION

UNIVERSITY

BS. (with honors)	Chemical Engineering University of California, Berkeley	1961
PhD	Chemical Engineering (minors: Chemistry & Mathematics) Oregon State University, Corvallis	1965

INDUSTRY (Selected Courses)

QA for Management and Research Staff
 Arctic Engineering, Univ. of British Columbia
 Principles of Project Management
 Principles of Safety Evaluation for Managers
 Principles of Management
 The Manager as Leader
 HAZWOPER 24 hour Training for Supervisors with 8 hour Refreshers
 RadWorker II Training (Hanford and Los Alamos National Laboratory)

PROFESSIONAL CERTIFICATIONS

Registered/licensed professional engineer in the States of Washington (#12231), Alaska (#EC 5925), Oregon (#17,054), Maryland (#21385), Arizona (#29767), and New Mexico (#13190)

Corrosion Specialist (#867) certified by the NACE Int'l (National Association of Corrosion Engineers)

Registered with The National Council of Examiners for Engineering and Surveying (#13634)

Held Dept. of Energy Q(3) and Dept. of Defense Secret clearances until mid-1991.

EXPERIENCE

— Current Responsibilities

In 1991, Dr. Divine was instrumental in organizing ChemMet, Ltd., PC, a licensed professional services engineering corporation for which he serves as Chief Engineer. He is in charge of the management of chemical and corrosion engineering tasks including environmental assessment efforts, evaluation of operational safety in industrial and nuclear facilities, and the development of programs which combine the principles of chemistry and materials. He is also responsible for assisting in the transfer of technology between governmental agencies and private industry.

JAMES R. DIVINE (continued)

Some of his recent projects involve the oversight of corrosion design evaluations for the US Army Corps of Engineers; evaluation and testing of water treatment processes; corrosion monitoring and evaluation of the safety of Hanford nuclear waste storage tanks and underground waste sites; the corrosion evaluation of waste and chemical processing operations; the evaluation of polymers for use at waste disposal sites; and the cathodic protection of fuel storage sites.

He serves as an Adjunct Faculty Member of the Chemical Engineering Department at the Tri-Cities Campus of Washington State University. He has taught courses in fluid flow, thermodynamics, and corrosion.

— Prior to 1991

Dr. Divine joined Battelle-Northwest in 1965 and was primarily concerned with studying corrosion mechanisms and kinetics in high-temperature water. He participated in programs aimed at establishing the effects of process parameters, including fluid hydraulics, heat flux, and radiation, on corrosion processes, corrosion product transport and deposition. Dr. Divine was also associated with studies on the dissolution of uranium and plutonium oxides, corrosion processes in nonaqueous solvent systems, and the electrodeposition of coatings on thin wires. During this period, he contributed to three invention reports and was a co-author of a US patent. He also developed, from a basic concept, a research program on corrosion of grinding steel in the mining industry that included international participants.

In 1974, he joined Westinghouse Hanford Company as a Senior Process Chemical Engineer in the development of the Acid Digestion Process for the reduction of combustible transuranic waste volumes. During this period he conceived of a novel method of processing acidic off-gases to reduce their effective corrosiveness which was prepared as an invention report, collaborated on the development of methods of waste volume reduction, and assisted in the preparation of Safety Analysis Reports.

He returned to Battelle-Northwest in 1978 as a Senior Research Engineer where he conducted studies on corrosion and the mass transport of corrosion products in aqueous systems as well as studies on chemical decontamination of nuclear reactor systems. He also consulted on refinery corrosion. As Technical Leader of the Electrochemical and Corrosion Processes Group, he had the added responsibility of monitoring the technical performance of a group of eight professionals while serving as project manager for his own programs. He guided activities to promote and market the capabilities of the group and section.

In 1983 - 1985, while serving as Technical Leader, he was promoted Staff Engineer. During this period, he was in charge of several technical programs as well as simultaneously serving in an administrative position. Typical programs included:

A corrosion evaluation program on storage tank construction materials in simulated Hanford caustic waste mixtures which included developing and evaluating methods for in-tank corrosion monitoring.

Development of inert anodes and cathodes for aluminum production by chemical and metallurgical engineering methods and by electrode reaction mechanisms studies using ac/dc methods.

Evaluation of atmospheric corrosion in Alaska for the purpose of extending the data base of the contiguous United States into the Cold Regions.

He served as Manager, Corrosion and Metallurgy Section for four years beginning in 1985. During this period, he was in charge of an average of 30 Exempt and Non-exempt staff, an average annual section funding of about \$5,000,000, a capital equipment inventory with a value of over \$6,000,000, and over 35,000 ft² of facility space.

He provided technical oversight on programs in the areas of: Corrosion Testing; High-temperature and High-pressure pH and Conductivity Sensor Development; Chemical Cleaning (Nuclear and Chemical Systems); Geothermal System Materials Monitoring; Hazardous Waste Barrier Development; DOE/Industry Technology Transfer; Operation of a 100-Unit Autoclave Facility; Basic Electrochemical Processes of Stress Corrosion Cracking; and Natural Gas Pipeline Corrosion.

JAMES R. DIVINE (continued)

Administratively he promoted the expansion of program development into new technical areas with the participation of all professional members of the section staff. He worked towards the simplification of the preparation of proposals, and instigated centralized control of Section Quality Assurance records to provide expeditious management oversight, increase staff acceptance to new regulations, and hold down costs. He developed and implemented a safety plan and training records system for the section that was copied for use at higher administrative levels. He also had developed and implemented an equipment inspection procedure for high temperature/pressure test equipment.

While Section Manager, Dr. Divine maintained his own technical activities where he consulted with corporate, national, and local groups, primarily in the area of corrosion and environmental effects on materials including the testing of improved clothing materials exposed to surety agents. He conducted studies as the principal investigator in these areas. A 15-20% level of effort was allocated to these technical efforts; during this period, he conceived of a method for analyzing impurities in motor oil for which an invention report was filed.

Following his tenure as Section Manager, he served as Principal Investigator and Project Manager for corrosion and materials test programs. Typical programs included studies on the corrosion of Hanford waste tanks and processing operations, the corrosion of steel in Hanford soil, and the testing and evaluation of polymeric liners for waste storage sites. He was also involved in technology transfer and the novel application of chemical engineering processes, as for example, when he conceived of, and was project manager, for a "novel" space suit glove developed by Battelle for NASA.

RECENT COURSES TAUGHT

Corrosion Short Course, with Dr. R. S. Johnson, at Idaho National Engineering Laboratory, July, 1994
Sessions on Toxicology and Confined Space Entry for HAZWOPER Course, 1996 & 1997
Session on Stoichiometry for PE Refresher course on Aug, 1998, WSU-Tri-Cities

PROFESSIONAL AFFILIATIONS

- American Association for the Advancement of Science, AAAS
- American Chemical Society, ACS
- American Institute of Chemical Engineers, AIChE
- American Water Works Association, AWWA
- American Society for Testing & Materials, ASTM
- ASM International
- Association of Consulting Chemists & Chemical Engineers
- National Association of Corrosion Engineers, NACE International
- National Society of Professional Engineers, NSPE
- Society of Plastics Engineers, SPE
- Society of American Military Engineers, SAME
- The Society of The Sigma Xi, Research Honor Society
- Tau Beta Pi, Engineering Honor Society
- Phi Lambda Upsilon, Chemistry Honor Society

KEY WORDS

chemical engineering arctic hazwoper hazardous radworker registered licensed
washington alaska oregon maryland arizona new mexico corrosion metal polymers
decontamination electrochemistry materials cathodic anodic protection sensors high-level
instruction research waste surety doe dod secret q(3) water radiation radioactive

PERSONAL

Commercial Pilot, Instrument rated; Airplane - single engine - land
Certified Flight Instructor - Airplane and Instrument
Civil Air Patrol: (Lt. Col.); past Chief Check Pilot, Eastern Washington
Member: National Association of Flight Instructors
Member, Tri-Cities Advisory Council on Literacy

JAMES R. DIVINE (continued)

PUBLICATIONS

1. Divine, J. R., and W. H. Nielson. 1967. *An Analysis of the Behavior of Arsenic in Hanford Reactor Systems*. BNWL-CC-1410, October.
2. Divine, J. R., and A. B. Johnson, Jr. 1971. *Characteristics of Crud Deposition and Its Influence on Fuel Rod Corrosion*. Proprietary Report, July.
3. Reid, W. T., G. Walker, H. Tannenberger, and J. R. Divine. 1972. *Planning Report on Fuel-Cell Substations for Urban Energy*. Internal Report Submitted to Battelle Memorial Institute, February.
4. FitzPatrick, V. F., J. R. Divine, and R. B. Richman. 1972. *Patent on Apparatus for the Detection and Measurement of Crevice Corrosion*. Patent No. 3,599,090, August.
5. Divine, J. R. 1972. *A Comparison of Stainless Steel and Zircaloy Coolant Samplers*. BNWL/JN-112, Proprietary Report, April.
6. Divine, J. R. 1973. *Waste Canister Longevity - Sea Bed and Polar Ice Environments*. Internal Report, March.
7. Divine, J. R. 1973. *Decontamination of Plutonia-Contaminated Thermal Reactor Systems*. Nuclear Technology 18:171-176, May.
8. Divine, J. R. 1974. *A Literature Survey on Nitrite Radiolysis*. Proprietary Report, January.
9. Divine, J. R. 1975. "Erosion" (Chapter 10) and "Sodium-Water Reactions" (Chapter 12) in: *A State-of-the-Art Review of Degradation Processes in LMFBR Materials, Volume II, Corrosion Behavior*. BNWL-1901, R. L. Dillon, Compiler, March.
10. Sikov, M. R., J. R. Divine, and P. L. Hackett. 1976. *Studies of Duplex (Copper-Coated) Wires for Use in Intrauterine Devices Contraception*. Vol. 13, January.
11. Divine, J. R. 1976. *A Method for Separating HCl and NO₂ From a Gas Stream*. Invention Disclosure to Westinghouse-Hanford.
12. Allen, C. R., J. R. Divine, and R. G. Cowan. 1977. *Safety Analysis Report for the Radioactive Acid Digestion Test Unit (RADTU)*. TC-893, January.
13. Perrigo, L. D. and J. R. Divine. 1978. *Impact of LWR Decontamination on Radwaste Systems*. PNL-SA-7381, January.
14. Hoernes, G. R., J. R. Divine, L. D. Perrigo, and L. G. Faust. 1979. *The Impact of Decontamination on Radioactive Waste Treatment Systems*. PNL-SA-7535, October. (Also Published in *Decontamination and Decommissioning of Nuclear Facilities*, MM Osterhout, Ed., Plenum Press, 1980.)
15. Divine, J. R., and L. D. Perrigo. 1980. *Decontamination Methods*. PNL-SA-7770, Rev. 1, Pacific Northwest Laboratory. Prepared for Presentation as Lecture at ASME Radioactive Waste Management Short Course, May 22.
16. Nelson, J. L., and J. R. Divine. 1980. *Hanford Transuranic Storage Corrosion Review*. PNL-3365, January.
17. Wick, O. J., J. R. Divine, and G. B. Dudder. 1981. *Steel Wear in Wet Grinding*, BN-SA-1004, Presented at the Society of Mining Engineers Fall Meeting, Tucson, AZ, October 17, 1979, February.
18. Divine, J. R., S. M. Bruemmer, A. B. Johnson, Jr., and R. H. Jones. 1981. *An Overview of Spent Fuel Pool Materials Studies at Pacific Northwest Laboratory*. PNL-SA-8807, Presented at CORROSION/81, Toronto, Canada, April 6-10, 1981.
19. Thompson, N. G., L. E. Vaele, J. R. Divine, J. L. Nelson, J. H. Payer, and T. J. Barlo. 1981. *Analysis of Complex Corrosion Conditions - Project B-234*, Report to Rockwell Hanford Operations by Battelle Memorial Institute, April.
20. Nelson, J. L., and J. R. Divine. 1981. *Decontamination Processes for Restorative Operations and as a Precursor to Decommissioning: A Literature Review*. NUREG/CR-1915, PNL-3706, April.

Xiangdong Feng

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509-373-7284 (office)
509-376-1638 (fax), E-Mail: X_feng@pnl.gov

Dr. Feng is a staff scientist at the Glass Development Group of Pacific Northwest National Laboratory (PNNL). Prior to that, he was a staff chemist at the Waste Glass Group of Argonne National Laboratory between 1991 and 1994. Dr. Feng also served as a Research Scientist at the Vitreous State Laboratory of the Catholic University of America from 1984 to 1991.

For the past fifteen years, Dr. Feng has been a leading scientist in the development of glass and ceramic products for nuclear and hazardous waste immobilization, and for industrial applications. Based on the current understanding of glass structure and glass chemistry, he developed thermodynamic models which have been utilized in optimizing glass and ceramic compositions to achieve the desired chemical durability, proper processing properties, and maximal loading of certain chemical constituents. He is also currently active in developing ceramic and organic composite materials that have wide applications in chemical separation, catalysis, controlled releases, and drug delivery.

His other interests and experience include:

- Glass and ceramic formulation, considering interfacial batch reactions, phase diagrams, and chemistry
- Elucidating the chemistry of glass and ceramic stability and its corrosion mechanisms
- Exploring the fundamentals of crystallization in a glassy matrix
- Developing test methods to evaluate the long-term performance of various glassy waste forms in vapor and in aqueous environments
- Utilizing and recycling wastes from petroleum, glass, and other industries for the production of marketable glassy products
- Designing and modifying the surface properties of glass and other inorganic materials at a molecular level to develop real time sensors and novel ion exchange materials for removing hazardous metals from wastewater; a new generation of water purification membranes; advanced biotreatment systems; antifouling filtration systems; electrorheological fluids; novel catalysts, and matrix for controlled release and delivery in agriculture and medical applications
- Kinetics studies of inorganic complexes.

Dr. Feng is currently a principal investigator and project manager for glass and ceramics development project for nuclear wastes; is leading a multi-institution and multi-nation project to develop the basic science on solubility and distribution of radionuclides in glasses and ceramics; a project to develop and demonstrate a ceramic-based Self-Assembled Monolayers on Mesoporous Supports (SAMMS) material for chemical separation, catalysis, and controlled release; and a project to develop polymer-coated ceramics for application in microtechnologies. His 1998 R&D budget is over \$1,400,000.

He has more than 90 scientific publications (including a recent article in *Science*), one book chapter, two patents granted, and five patents in application. He has served as a panelist and committee member for many workshops and technology committees on glass waste form development, composition optimization, glass property modeling, and testing method

development. He has also served as organizing committee member and session chair for numerous international technical meetings and symposiums on glasses and ceramics. He is a member of the American Chemical Society, the Materials Research Society, and the American Ceramic Society.

EDUCATION AND WORK HISTORY:

August 1994 to present:	Principal Investigator and Senior Research Scientist, Pacific Northwest National Laboratory, Richland WA
March 1991 - July 1994:	Principal Investigator and Chemist, Argonne National Laboratory, Argonne, IL
May 1988- February 1991:	Group Leader and Research Scientist, Vitreous State Lab, Catholic University of America, Washington, D.C.
September 1982 - April 1988:	Research Assistant, Vitreous State Lab, Catholic University of America, Washington, D.C.
Ph.D., Inorganic/Physical Chemistry, 1988: Catholic University of America, Washington, D.C.	
M.S., Inorganic/Physical Chemistry, 1984: Catholic University of America, Washington, D.C.	
B.S., Chemistry, 1980: Hunan Normal University, China.	

HONOR AND AWARDS

Outstanding Performance Award of Pacific Northwest National Laboratory (1997)
Nominated for 1998 Discover Award
Nominated for 1998 R&D 100 award
Who's Who in Science and Engineering (4th edition, 1998)
Honorary Professor of Hunan Normal University Hunan Normal University (1994)
Honorary Professor of the China Institute of Atomic Energy (1994)
National Graduate Scholarship from the Ministry of Education of China (1982)

RECENT PATENTS and PATENT APPLICATIONS

X. Feng and R. E. Einziger, "A Method for Converting Spent Nuclear Fuels and Surplus Plutonium into Vitreous Ceramic Final Waste Form," U.S. Patent # 5656009, August 1997

X. Feng and R. E. Einziger, "Ceramics for Immobilizing Spent Nuclear Fuels and Surplus Plutonium," U.S. Patent 5597516, February 1997.

X. Feng, Jun Liu, and Glen E. Fryxell, "Self-assembled Mercaptans on Mesoporous Silica (SAMMS) for Mercury Separation and Stabilization," U.S. Patent Application filed February 7, 1997, PNNL #E-1479.

X. Feng, and Jun Liu, "Molecular Filters for Ultrahigh Purity Water at Low Cost," PNNL Invention Report filed January 12, 1996, PNNL #E-1464.

X. Feng, John G. Darab, P. R. Hrma, Y. Peng, and Peter A. Smith, "A Low-temperature Waste Form and a Low-temperature Process for Immobilizing Technetium," PNNL Invention Report filed January 16, 1997, PNNL #E-1552.

X. Feng and Jun Liu, "A Smart Membrane System for Antifouling in Water Filtration/Treatment and for Microdevices," PNNL Invention Report filed May 30, 1997, PNNL #E-1601.

G. E. Fryxell, X. Feng, and J. Liu, "Microporous Materials with High Permeance and Selectivity Based on Mesoporous SAMMS," PNNL Invention Report filed June 9, 1997, PNNL #E-1607.

J. M. Perez, X. Feng, and J. Liu, "SAMMS Materials for Removal of Tritium from Water-containing Oil Wastes," PNNL Invention Report filed June 13, 1997, PNNL #E-1606.

J. R. Bontha and X. Feng, "Tritium Separation From Water through Formation of Clathrate Hydrate Nucleated on the Tritiated Small Alkane Chains Within the Pores of SAMMS," PNNL Invention Report filed June 13, 1997, PNNL #E-1626.

Guy Maurin

Current Position: Manager of the Technology Department (Technical Division of SGN). NHC is a subsidiary of SGN and COGEMA.

Education: Graduate Engineer from the Ecole Polytechnique Paris and ENSTA (Nuclear Engineering)

Languages: English, French and German

Experience:

SGN: Nuclear Fuel Cycle Engineering

1997: Manager of the Technology Department
In charge of physical processes and equipment, control, instrumentation and electricity, robotics, welded equipment, civil works, materials, inspection.

1991: Manager of Mechanical, Welded Equipment and Scientific Calculations Department

1990: Manager of Scientific Calculations.

1989: Deputy Manager of Process Department, Responsible for the Zirconium Project.

1986-1988: Manager of the Safety Department for the JNFS Reprocessing Plant Project (Japan)

NOVATOME: Fast Breeder Reactors Engineering

1984-1986: Manager of the Safety and Reliability Department.

1982-1984: Manager of the Safety Analysis Group.

1980-1982: Manager of the Accident Analysis Group.

1976-1980: Detached to CEA Cadarache.
Development of calculation programs, including thermodynamic, neutronic, hydrodynamic, and mechanical modules.

DONALD R. OLANDER

Curriculum Vitae

Education: AB (Chemistry) Columbia University (1953)
BS (Chemical Engineering) Columbia University (1954)
ScD (Chemical Engineering) Massachusetts Institute of Technology (1958)

Awards and Fellowships:

Euratom Fellowship (1965-66)

FOM-Institute of Atomic and Molecular Physics Fellowship (1972-73)

Award for Outstanding Contribution to Nuclear Technology from the Materials Science and Technology Division of the American Nuclear Society (1976)

Invited Lecture Tour, People's Republic of China, June 1982

Fellowship of the Japan Society for the Promotion of Science for lecture tour, May 1983

Sustained Outstanding Research Award in Materials Chemistry, USDOE Office of Basic Energy Research 1983

Fellow of the American Nuclear Society, 1984

National Science Council (Taiwan) Lectureship in Nuclear Engineering (1987)

Lady Davis Visiting Professor, Technion, Israel, 1987-88

Present Position:

Professor, Department of Nuclear Engineering, University of California, Berkeley

Senior Faculty Scientist, Materials Science Division, Lawrence Berkeley Laboratory

Research Interests: Materials chemistry of nuclear fuels

Professional Activities:

Editorial Boards - Journal of Nuclear Materials; High Temperature Science

TMI-2 Accident Evaluation Panel, Idaho National Engineering Laboratory, (EG&G)

Chairman, Review Committee for the Experimental Breeder Reactor II Project (University of Chicago)

Steering Committee for Workshop on High Temperature Processes in Severe Reactor Accidents (National Academy of Sciences, U. S. NRC)

U.S. Organizer, U.S. Japan Joint Seminar on High-Temperature Chemistry in Light-Water Reactors, (NSF-DOE)

Chairman of Panel on Reactor Fuel Technology, Workshop on Transactinium Science (LLNL)

Chairman, Review Committee on NPR-MHTGR Fuel Development Program Plan (EG&G)

Advisory Council, Institute for Transactinium Science (LLNL)

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Section 3 of 3

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5.0 SHIELDING EVALUATION

5.1 SHIELDING DESIGN DESCRIPTION

The multi-canister overpack (MCO) is designed to ensure confinement and criticality control of radioactive materials. The MCO also provides shielding for worker protection only immediately above the top plug where process operations occur beyond the protection of any additional facility shielding materials. This chapter describes the radiation source shielding analysis performed on the shield plug and the results of the analysis. A more detailed description of the MCO shield plug shielding analysis is provided in a separate document (HNF 2000). The analysis calculation files are maintained by the shielding evaluation contractor, Parsons Infrastructure and Technology, Inc. Note that side radiation emissions that do not backscatter into the MCO are not a significant contributor to the dose rate above the shield plug and are not evaluated in this analysis. However, side radiation is considered in the evaluation of the MCO handling machine shielding provided in a separate analysis (ALSTOM 1998).

5.2 RADIATION SOURCE DEFINITION

The selection of the bounding "shielding design basis" fuel considered the range of radionuclide compositions in N Reactor fuel (i.e., Mark IV, Mark IVB, and Mark IA) and single pass reactor fuel predicted to exist within the K East and K West Basins at the Hanford Site. These fuel compositions are based on varying exposures and decay periods reported for each key (or group of fuel assemblies) in the accountability database (Praga 1998a). The selected shielding design basis feed shown in Section 2.2.6.2, Table 2-6 (see also Section 3.4 and Table 3.8 of HNF-SD-SNF-TI-009, *105-K Basin Material Design Basis Feed Description for Spent Nuclear Fuel Project Facilities, Volume 1, Fuel* [Praga 1998a]) represents a feed with the highest gamma shielding requirements as compared to the nominal feed. The isotope ^{137}Cs is the dominant gamma contributor in either case (i.e., highest feed versus nominal feed). The shielding design basis fuel material was selected to include the fuel with the highest ^{137}Cs content because of its relative abundance in the fuel and its significant gamma radiation. This feed is used to evaluate gamma shielding provided by the plant structure and equipment to determine whether potential peak radiation dose rates are acceptable (see Table 1.1 of HNF-SD-SNF-TI-009 [Praga 1998a]). This feed represents material that would be in the facility for short campaigns and would not be used for evaluations of performance over the entire facility life.

The key or fuel group with the highest actual cesium content is Mark IV fuel with a ^{240}Pu content of 15.74% discharged from the reactor in 1984. For conservatism, Mark IV fuel (0.95% ^{235}U) irradiated to 16% ^{240}Pu content (i.e., 6.5% higher than maximum fuel in inventory), and aged 13.5 years to May 31, 1998 (Praga 1998a) was selected as the shielding design basis. The selected shielding design basis fuel allows any combination of fuel canisters that may be encountered within the K Basins to be loaded into an MCO without concern for exceeding shielding requirements to limit peak (as opposed to average) dose rates. The photon (e.g., gamma rays) source term was expected to be the major shielding concern compared to the

neutron source term, but the maximum neutron source term was also evaluated because the neutron source may not necessarily be bounded by the shielding design basis source shown in Table 2-6.

Appendix H of HNF-SD-SNF-TI-009 (Praga 1998a) shows the neutron source for N Reactor fuel (i.e., Mark IV and Mark IA) from the ORIGEN2 calculational data as a function of ^{240}Pu burnup and discharge time (PNNL 1997). Note that the higher burnup fuels (16% ^{240}Pu) provide the highest neutron source. However, there is no Mark IA fuel at the 16% ^{240}Pu burnup level. The maximum ^{240}Pu content in Mark IA fuel is 14.4% (see Appendix A of HNF-SD-SNF-TI-009 [Praga 1998a]). Therefore, only the 16% ^{240}Pu Mark IV fuel was considered as the highest neutron source fuel for the MCO dose analysis.

It should be noted that the ORIGEN2 calculations inherently assume an oxide fuel and provide calculation of the (α, n) contribution to the neutron source (see Appendix H of HNF-SD-SNF-TI-009 [Praga 1998a]). Because N Reactor fuel is a metal fuel and not an oxide fuel (Reilly 1998), the (α, n) contribution will be included in the source term to conservatively bound the small amount of fuel oxidation expected and the presence of any light trace elements or impurities. The data in Appendix H of HNF-SD-SNF-TI-009 (Praga 1998a) indicate that the (α, n) contribution is 38% of the total neutron source contribution for the 16% ^{240}Pu Mark IV fuel at a discharge time of 10 years and 45% at a discharge time of 20 years.

Appendix H of HNF-SD-SNF-TI-009 (Praga 1998a) provides the activity ratios of ^{244}Cm to ^{137}Cs for various N Reactor burnup and discharge times based on the ORIGEN2 calculational data. The data in Appendix H of HNF-SD-SNF-TI-009 (Praga 1998a) show that the 16% ^{240}Pu burnup fuel has the highest neutron-to-photon ratio. Again, this result supports using the 16% ^{240}Pu Mark IV fuel as the neutron source fuel. Similarly, this same fuel was previously chosen as the shielding design basis fuel based on its high ^{137}Cs content and photon source (Praga 1998a) and was used in the MCO shield plug photon dose analysis.

A very small amount of Mark IV fuel has 16.7% ^{240}Pu burnup, but it is older, with a discharge time of approximately 20 years. The small increase in burnup and the longer decay time yields a few percent less ^{244}Cm inventory but a few percent more ^{240}Pu inventory, thus making an overall small change in the neutron source. This small change is not considered significant with respect to dose rate calculations and is considered within the calculational uncertainties. This older fuel also has approximately 15% less ^{137}Cs inventory on a per metric ton of uranium (MTU) basis compared to the 16% ^{240}Pu burnup fuel. Thus, it is not an appropriate fuel source to use for the photon source basis. Furthermore, because it is reasonable to choose the same basis fuel for both neutron and photon sources, the 16% ^{240}Pu Mark IV fuel becomes the choice of fuel source.

The percentage of high burnup fuel with 16.7% ^{240}Pu is small compared to the total. Only approximately 9 MCOs out of a total of approximately 400 would be needed to accommodate this high burnup fuel assuming an MCO fuel loading of 50% high burnup (i.e., 16.7% ^{240}Pu) fuel with other burnup fuels. These items suggest that using the 16.7% ^{240}Pu burnup fuel as the shielding design basis is not appropriate when considering the small inventory and processing time commitment.

There are two different sets of MCO basket designs: one set of fuel and scrap baskets for Mark IV fuel and one set of fuel and scrap baskets for Mark IA fuel. The Mark IA baskets are shorter, with a larger center tube, than the Mark IV baskets. Therefore, the Mark IA baskets have a smaller capacity and would contain less fuel. The scrap basket loading will be controlled by administrative criticality limits (Pajunen 1997). The criticality mass limit established for a Mark IV fuel scrap basket is 980 kg, and the criticality mass limit established for a Mark IA fuel scrap basket is 575 kg (Kessler and Peck 2000). These limits are related to K Basin pool loading activities, and both Mark IV and Mark IA fuel sources were analyzed to ensure the dose calculation results were conservative (see discussion in Section 5.4.3). Modeling of the MCO scrap basket assumed loading at the criticality mass limits for conservatism. This fuel mass was homogenized over the basket volume to facilitate the dose calculation.

The MCO loading configuration used for the shielding design basis fuel consisted of 6.34 MTU in Mark IV fuel elements irradiated to 16% ^{240}Pu content. The fuel was contained in a five-tier MCO (i.e., 54 Mark IV fuel assemblies per tier, for a total of 270). A very thick shield, simulating the cask wall, was used on the sides and bottom of the MCO model to maximize backscatter. The side radiation emissions that do not backscatter into the MCO are not a significant contributor to the dose rate above the shielding plug. The majority of the photon dose above the shield plug was shown to be from the top tier of fuel elements (see Section 5.3.1).

5.2.1 Photon Source

The photon source spectra were calculated based on the Brookhaven National Laboratory database (BNL 1996) and the shielding design basis source term given in Section 2.2.6.2, Table 2-6, and HNF-SD-SNF-TI-009 (Praga 1998a). Table 5-1 lists the fraction of the photon source per MCO distributed in 11 energy bins. Each bin value represents the upper bound of the energy bin. The total photon strength is 8.29×10^{15} photons/s/MCO.

A significant portion of the photon activity that contributes to the dose beyond the shield is from the $^{137\text{m}}\text{Ba}$ decay associated with ^{137}Cs . The primary energy associated with this decay of 0.662 MeV is overestimated by the 0.6 MeV to 0.8 MeV binning structure. The effect of using this higher energy results in overestimating the energy flux entering the detector and possibly, to a lesser extent, underestimating the fraction of the energy deposited in the detector. The effects of this binning structure were estimated in the analysis and showed a variation of the dose by a factor of approximately 1.5 to 3.0. This factor is based on using the Monte Carlo N-particle (MCNP) code to analyze a solid iron cap with a density 8.03 g/cm^3 and then evaluating the dose for 0.662 MeV photons and the original energy bin for the same source term. Although this conservatism could be removed by dividing the results by slightly less than this factor, it has not been changed as part of the conservatism of the analysis. This conservatism should offset the presence of void space in the cask metal and unfilled portions of the drilled areas in the shield plug introduced as part of the manufacturing process.

Table 5-1. Fraction of Photons in Various Energy Groups.

Upper energy bound (keV)	Fraction
100	0.089
200	0.206
300	0.0032
400	0.00037
500	0.19
600	0.0047
800	0.496
1,000	0.0031
1,500	0.0059
2,000	0.00076
2,500	0.0000083

For dose rates above the MCO plug, the upper 10 cm of the top tier were treated as a separate source and analyzed in detail (i.e., all fuel elements). Only representative fuel elements were selected for analysis below the top 10 cm of the first tier. The bottom four tiers of fuel elements were treated as single fuel elements and have negligible impact on the results because of self-shielding effects. This is based on a comparison of dose rates associated with the top 10-cm of the fuel layer and the remainder of that fuel layer.

Figure 5-1 provides a depiction of the source regions and is discussed in Section 5.3.1. A comparison of dose rates at the bottom of the MCO plug indicates that the top 10 cm of source contribute about 96% of the total photon dose. Indeed, an independent shield plug dose analysis calculation of the photon dose rates at the bottom of the MCO plug determined that the top 10 cm of source contribute 98% of the total photon dose above the MCO shield plug for maximum or average sources (Hillesland 1997).

5.2.2 Neutron Source

The neutron source term was calculated using ORIGEN2 (Wittekind 1994b) and the shielding source term given in the Shielding Design Bases provided in HNF-SD-SNF-TI-009 (Praga 1998a) (see the discussion in Section 5.2). The neutron source term is given in Section 2.2.6.2, Table 2-6. Neutron source geometry was determined using all baskets. The source addresses both spontaneous fission and (α ,n) reactions as summarized in Table 5-2.

The (α ,n) contribution was calculated assuming an oxide fuel. Because N Reactor fuel is not an oxide fuel (Praga 1998a), the (α ,n) contribution conservatively applies to that portion of the fuel that has interstitial water, oxygen (e.g., oxides of uranium), and various other materials from structural failure, corrosion, and impurities of manufacture. Thus, the addition of the spontaneous fission results and the (α ,n) contribution would clearly bound the results. The spontaneous fission results are projected to be a reasonable estimate for typical fuel in good condition. With 6.34 MTU per MCO, this results in a total neutron source strength of 7.3×10^6 neutrons/s/MCO (i.e., spontaneous fission and [α ,n] contributions).

Table 5-2. Maximum Neutron Source Term for the Multi-Canister Overpack.

Source component source	Source strength (neutrons/s/MCO)
(α ,n)	2.9 E+06
Spontaneous fission	4.3 E+06
Total	7.3 E+06

MCO = multi-canister overpack.

Appendix H of HNF-SD-SNF-TI-009 (Praga 1998a) provides an evaluation of the neutron source term for use in the shielding design basis. Based on the selection of 16% ^{240}Pu Mark IV fuel for the neutron source term, Appendix H of HNF-SD-SNF-TI-009 (Praga 1998a) shows the contribution of individual isotopes to the neutron source term by percentage. The (α ,n) reaction is dominated by the following isotopes:

- For the 10-year discharge case
 - ^{238}Pu (20%)
 - ^{239}Pu (25%)
 - ^{240}Pu (17%)
 - ^{241}Am (37%)
- For the 20-year discharge case
 - ^{238}Pu (15%)
 - ^{239}Pu (21%)
 - ^{240}Pu (14%)
 - ^{241}Am (49%)

The spontaneous fission reaction is dominated by the following isotopes:

- For the 10-year discharge case
 - ^{240}Pu (54%)
 - ^{244}Cm (39%)
- For the 20-year discharge case
 - ^{240}Pu (62%)
 - ^{244}Cm (31%).

As discussed in Section 5.2, reasonable limiting case assumptions were used to select a reasonable neutron source term. Based on the data, the 16% ^{240}Pu burnup Mark IV fuel was used for most shielding calculations. The Mark IA fuel also was considered to ensure conservatism in evaluating the impact of the scrap basket. These fuels were used to select the applicable energy spectra for calculation. Variations in dose for different fuel types and burnup levels can be estimated by linearly adjusting the dose results based on weighting with ^{137}Cs activity. The associated doses will change linearly with total source activity assuming the energy distribution and geometry are unchanged. Thus, a bounding value for other projected fuel loads can be projected based on the results of this analysis.

The Monte Carlo watt spectrum function for ^{244}Cm (Breismeister 1997) was used as a representative energy spectrum function to represent the different isotopes. This is considered reasonable because of the small differences among the energy shapes for the spontaneous fission spectra of the different isotopes as shown in Figure 5-14. Comparing the spontaneous fission energy shapes for ^{240}Pu and ^{244}Cm shows that the average neutron energy is higher using the ^{244}Cm spectrum (i.e., 2.15 MeV for ^{244}Cm and 1.98 MeV for ^{240}Pu [Breismeister 1997]), thus making the choice of using the ^{244}Cm spontaneous fission energy shape conservative (i.e., the effect of downscatter would conceivably place more neutrons in the energy levels important to dose rates above the shield plug). This function is shown below.

$$f(E) = C \exp(-E/.906) \sinh([3.848E]^{0.5})$$

where E is the neutron energy in MeV and C is a constant to normalize the integral over $f(E)$ to unity. The fission spectrum for ^{244}Cm , which has an average neutron energy of 2.15 MeV (Breismeister 1997), is given in Table 5-3. The energy distribution of the (α ,n) neutrons is given in Table 5-4 as predicted by ORIGEN2 for the fuel mixture; the average neutron energy of this distribution is 2.01 MeV (Jobs and Liskien 1983). Induced fission neutrons are generated by the MCNP calculation (Breismeister 1997).

Table 5-3. Energy Distribution of Neutrons from Spontaneous Fission Events
Based on the Curium-244 Spectrum. (2 sheets)

Upper energy (MeV)	Cumulative probability	Probability of bin
0.00	0.00	0.00
0.10	0.01194	0.00194
0.20	0.03167	0.01972
0.30	0.05590	0.02424
0.40	0.08316	0.02726
0.50	0.11250	0.02934
0.60	0.14326	0.03076
0.70	0.17493	0.03167
0.80	0.20711	0.03218
0.90	0.23950	0.03238
1.00	0.27182	0.03232
1.20	0.33550	0.06368
1.40	0.39690	0.06140
1.60	0.45523	0.05832
1.80	0.50997	0.05475
2.00	0.56087	0.05090
2.20	0.60782	0.04695
2.40	0.65083	0.04301
2.60	0.69001	0.03917
2.80	0.72550	0.03549
3.00	0.75752	0.03202
3.20	0.78628	0.02876
3.40	0.81202	0.02574
3.60	0.83499	0.02297
3.80	0.85542	0.02043
4.00	0.87353	0.01812

Table 5-3. Energy Distribution of Neutrons from Spontaneous Fission Events
Based on the Curium-244 Spectrum. (2 sheets)

Upper energy (MeV)	Cumulative probability	Probability of bin
4.20	0.88956	0.01603
4.40	0.90371	0.01415
4.60	0.91616	0.01245
4.80	0.92711	0.01095
5.00	0.93671	0.00960
5.50	0.95578	0.01907
6.00	0.96931	0.01354
6.50	0.97884	0.00953
7.00	0.98550	0.00665
7.50	0.99011	0.00461
8.00	0.99329	0.00318
9.00	0.99695	0.00366
10.00	0.99863	0.00169
11.00	0.99940	0.00076
12.00	0.99974	0.00034
13.00	0.99989	0.00015
14.00	0.99995	0.00007
15.00	0.99998	0.00003

Table 5-4. Energy Distribution of Neutrons from (α ,n) Events as Predicted by ORIGEN2 for the Fuel Mixture. (2 sheets)

Upper energy (MeV)	Cumulative probability	Probability of bin
0.00	0.00	0.00
0.10	0.01059	0.01059
0.20	0.02243	0.01184
0.30	0.03396	0.01153
0.40	0.04766	0.01371
0.50	0.06636	0.01869
0.60	0.08738	0.02103
0.70	0.11044	0.02305
0.80	0.13567	0.02523
0.90	0.15981	0.02414
1.00	0.17975	0.01994
1.10	0.20062	0.02087
1.20	0.22321	0.02259
1.30	0.24860	0.02539
1.40	0.27601	0.02741
1.50	0.30405	0.02804
1.60	0.33349	0.02944
1.70	0.36542	0.03193
1.80	0.40093	0.03551
1.90	0.43785	0.03692
2.00	0.47664	0.03879
2.10	0.51558	0.03894
2.20	0.55623	0.04065
2.30	0.59751	0.04128
2.40	0.63707	0.03956
2.50	0.67492	0.03785

Table 5-4. Energy Distribution of Neutrons from (α ,n) Events as Predicted by ORIGEN2 for the Fuel Mixture. (2 sheets)

Upper energy (MeV)	Cumulative probability	Probability of bin
2.60	0.71137	0.03645
2.70	0.74611	0.03474
2.80	0.77819	0.03209
2.90	0.80935	0.03115
3.00	0.83863	0.02928
3.10	0.86449	0.02586
3.20	0.88879	0.02430
3.30	0.90966	0.02087
3.40	0.92664	0.01698
3.50	0.94097	0.01433
3.60	0.95327	0.01231
3.70	0.96324	0.00997
3.80	0.97181	0.00857
3.90	0.97928	0.00748
4.00	0.98536	0.00607
4.10	0.99081	0.00545
4.20	0.99439	0.00358
4.30	0.99720	0.00280
4.40	0.99891	0.00171
4.50	1.00000	0.00109

5.3 SHIELDING MODEL SPECIFICATION

5.3.1 Configuration of the Shielding and Source

Although this discussion is limited to the immediate vicinity on top of the shielding plug, a reflector, equivalent to a cask, is included in the model to prevent a potential underestimation of dose rates caused by backscattering of photons into the fuel matrix by the cask walls. There also is a potential for overestimation from air scatter back into the detectors from radiation passing beyond most or all of the shield plug.

The neutron dose is estimated based on the result of doses calculated for localized detectors using MCNP (see Section 5.4.2.2)

The modeling for the dose calculations and shielding is based on the MCO characteristics described in detail in Section 1.2 of this report and summarized as follows.

- The reflector and MCO are made of stainless steel.
- The MCO shield plug is made of stainless steel.
- The inside diameter of the MCO is 23 in., and the inside height is 148 in. to the bottom of the MCO plug.
- The MCO plug is 10.5 in. thick.
- The MCO sidewall is 0.5 in. thick, and the reflector sidewall is 8 in., for a total thickness of 8.50 in.
- The bottom of the MCO is 2.0 in. thick, and the bottom of the reflector is 8 in. thick, for a total bottom shield thickness of 10.0 in.

Figure 1-1 shows the MCO with the shield plug in place and the cover over the MCO plug. The cover is not in place for loading and drying activities. The drawing shows the five-basket configuration. Figure 5-2 provides details of the MCO shield plug monolith. The bolt penetrations were assumed to contain bolts, so they were not included in the MCO model of the monolith and not included in the MCNP model of the MCO shield plug. Where sloped or curved surfaces occurred, the MCNP model of the shield plug assumed worst material (i.e., the material that provides the least shielding impact and thus allows the greatest contribution to photon dose) and modeled these surfaces as straight lines. The cask-MCO interface detail was used as the basis for the MCNP model of the monolith with gaps between mating surfaces assumed to be 0.0625 in.

Figure 1-8 shows the penetrations in the MCO shield plug that were used in the shielding model. The structures beneath the shield plug, shown in Figure 1-1, sheet 2, also were included in

the MCNP model, but the filter materials in these structures were not included in the model. It was assumed in the MCO modeling that the valved openings will be filled with a process valve mechanism that reduces the penetrations in the upper portion of the MCO shield plug to 1-in.-diameter holes. The data used in modeling the MCO shield plug as described above are summarized in Figure 5-3.

The baskets were modeled as 3-in.-thick disks of steel with holes the size of a fuel assembly diameter. The fuel elements were assumed to be two concentric cylindrical shells of uranium encased in steel, consistent with Mark IV fuel as shown in Figure 5-4. The fuel elements were assumed to be located as shown in Figure 5-5 depending on fuel type. The central cylinder was assumed to be a cylindrical shell of steel containing an air void throughout its entire length. There are 54 Mark IV fuel elements in the relative positions shown in Figure 5-5. (Note that position 28 in this matrix is the void central cylindrical shell.) Figure 5-6 shows samples of the MCNP model geometry and are discussed in Section 5.4.1.

As shown in Figure 5-1, the spent nuclear fuel inside the MCO was modeled in three sections: the first 10 cm of top tier fuel, the balance of the top tier layer of fuel, and the remaining four lower tiers of fuel. This was based on results of previous calculations that determined that the upper 10 cm was the major contributor to the photon dose with the balance of the first layer of fuel providing a majority of the remaining dose received above the MCO shield plug. The first 10 cm of top tier fuel contributed 96% of the photon dose, and the balance of the top tier layer of fuel contributed approximately 4% of the photon dose. Thus, the remaining four tiers of fuel contributed less than 1% and were not considered further for photon dose analysis.

The calculations treated each of the 54 fuel locations as a separate source with each of the two cylindrical shells considered separately. The results were then consolidated to estimate the dose above the MCO shield plug for the top layer. For the second layer, calculations were based on a sample (i.e., 10 elements) of the 54 element doses with emphasis on those fuel elements below penetrations. Each of the fuel elements was assigned a dose value based on the most similar location (or average of similar locations) in the sample calculated, and then these doses were consolidated with the doses for the first layer. A similar process was applied to the remaining layers of fuel elements with a smaller number of samples; however these layers did not contribute significantly to the dose (i.e., their contribution was much less than 1%).

The scrap basket portion of the analysis was completed by taking an existing MCNP input deck and adding the structure of the basket. This was done by constraining the outer diameter of the basket to fit within the MCO and the inner diameter to the existing hole size. The basket bottom plate thickness was based on Figure 1-6 and is made of steel with a density of 8.03 g/cm^3 . The basket sides consisted of copper material with a density of 8.96 g/cm^3 and with dimensions taken from Figure 1-6. The source for the scrap basket was a homogenized steel and uranium mixture, which filled the entire annular volume of the basket, with a reduced density to reflect the presence of air-filled void spaces within this material. The fuel content for the basket (i.e., the source term for this basket) was assumed to be the criticality limit for fuel material for the applicable fuel types. Mark IV fuel and Mark IA fuel were evaluated. These two fuel types yielded very different densities — 9.40 g/cm^3 for the Mark IV fuel and 5.52 g/cm^3 for the

Mark IA fuel. Only a photon source was considered for the scrap basket, and the neutron source was determined based on the photon-to-neutron ratio found in the MCO analysis.

5.3.2 Material Properties

Shielding consists of the reflector, retaining collar, MCO, and MCO shield plug, all made of 304L stainless steel. During fabrication, the parts will be forged, ultrasonically inspected, and dye-penetrant examined after machining. Any porosity will be apparent. The appropriate materials (i.e., air, steel, uranium) were included in the appropriate locations and surrounding areas to ensure that the scattered radiation was not underestimated. Table 5-5 summarizes the material properties. The shielding attenuation properties are obtained from the data library for the MCNP computer code (Breisemeister 1997). When the shielding analyses were initiated, a specific steel density was provided, but the type of steel was not specified. Thus iron was used to represent steel in all calculations in these analyses. Based on specific calculation runs for photons and neutrons, representing steel with iron resulted in a minor change in the results, specifically, <3% for photons and approximately 7% for neutrons.

5.4 SHIELDING ANALYSES

5.4.1 Computer Programs

The Monte Carlo computer code, MCNP (Breisemeister 1997), was used to perform the dose rate calculations. MCNP has powerful geometry routines and uses an ENDF/B database for cross sections. The MCNP quality assurance verification and validation documentation for this task maintained by the design agent who conducted the verification and validation of the MCNP Version 4A and 4B software for the dose calculations (McConn and Woolfolk 1996). The program and associated documentation of the design agent's verification and validation have been audited satisfactorily.

Figure 5-6 illustrates a sample of the MCNP geometry used in these analyses for the MCNP modeling. In the modeling, MCNP distinguishes among areas containing uranium, air, steel, International Commission on Radiation Units and Measurement (ICRU) tissue, and the balance of the universe that is not considered in the analysis. See Figures 5-2, 5-3, and 5-5 for more detailed data on the design and dimensions assumed for the MCO. Note that this analysis only evaluates the effectiveness of the MCO shield plug; the cask's steel walls are not intended to be representative of the designed cask wall but are intended to function as a backscatter shield for the calculations. Based on the doses calculated for the top 10 cm of fuel and the balance of the top layer of fuel for the areal dose calculation, the contribution for most of the lower layers of fuel (i.e., below 10 cm) in the cask is not significant in calculating the photon dose because of the shielding of the source material above the lower layers.

Table 5-5. Materials and Densities Used for Shielding.

Material	Density (g/cc)	Remarks
Steel	8.03	Used in the reflector, collar, MCO, and MCO plug*
Air	0.001293	Provides scattering media for neutrons; photon models neglected attenuation and scattering in air out to 6 m
Uranium	17.86	Fuel elements modeled as ^{238}U with 0.947% ^{235}U ; also, assembly end caps conservatively replaced by uranium (source)
Tissue	1.0	Detectors

*These items will be fabricated of 304L stainless steel. Based on the nominal density for 304 stainless steel obtained from the *Handbook of Stainless Steels* (D. Peckner and I. M. Bernstein, 1977, McGraw-Hill, Incorporated, New York, New York), a value of 8.03 g/cm³ (0.29 lb/in³) was chosen for this shielding analysis.

MCO = multi-canister overpack.

The neutron cross sections are provided with the MCNP software. Table 5-6 summarizes the neutron cross sections used in the analysis consistent with the applicable material temperature. The most current applicable verified neutron cross sections (i.e., released by Los Alamos National Laboratory through the Radiation Safety Information Computational Center) available were used. For consistency with the other material an ENDF60 series cross section was chosen for iron although natural iron was not available in this library. Because natural iron is composed mostly of ^{56}Fe (approximately 92% of natural iron), this specific cross section was used for natural iron.

The photon cross sections used in the calculations are stored in the xsdir file in the MCNP Code, Version 4B (Breisemeister 1997), installed as indicated in the instructions provided by the Radiation Shielding Information Computation Center (Breisemeister 1997). The photon spectrum used was for the natural isotope of the material from the ZAID=ZZZ000.01P library (i.e., MCNP library mcplib1). Cross section data for elements 84, 85, 87, 88, 89, 91, and 93 as indicated in the MCNP manual are from LA-3753, *Photon Cross Sections from 0.001 to 100 MeV for Elements 1 Through 100* (Storm and Israel 1967), but none of these elements were used. The cross section data used in this problem, as indicated in the MCNP manual, are based on data from ENDF (Hubbell et al. 1973) for energies from 1 keV to 100 MeV. Fluorescence data as indicated in the MCNP manual are from LA-5240-MS, *MCNP Code Fluorescence-Routine Revision* (Everitt and Cashwell 1973).

Table 5-6. Neutron Cross Sections Used in the Multi-Canister Overpack Shield Plug Analysis.

ZAID	Element	Cross section library	Temperature (K)	Date
1001.60c	Hydrogen	endf601	294	1989
LWTR.O1T ^a	Hydrogen	Thermal S (α, β)	300	NA
6000.60c	Carbon	endf601	294	1989
7014.60c	Nitrogen	endf601	294	1992
8016.60c	Oxygen	endf601	294	1990
11023.60c	Sodium	endf601	294	1977
12000.60c	Magnesium	endf601	294	1978
15031.60c	Phosphorous	endf601	294	1977
16032.60c	Sulfur	endf601	294	1977
18000.59c ^b	Argon	misc5xs1	294	1982
19000.60c	Potassium	endf601	294	1974
20000.60c	Calcium	endf601	294	1980
26056.60c	Iron	endf601	294	1989
92235.60c	Uranium-235	endf601	294	1989
92238.60c	Uranium-238	endf601	294	1993

^aThis cross section is a special physics treatment for low energy neutrons (<4 eV) to account for molecular interactions.

^bBreismeister, J. F., Editor, 1997, *MCNP — A General Monte Carlo N-Particle Transport Code, Version 4B*, LA-12625-M, Los Alamos National Laboratory, Los Alamos, New Mexico.

The MCNP program (Breismeister 1997) uses the natural probabilities that various events occur (e.g., collision, fission, capture). The particles are followed from event to event with the next event determined by randomly sampling from the number of possible events according to the natural event probabilities. Independent probability distributions specified by the user or based on built-in functions are used for the source variables of energy, time, and position and direction, and for other parameters such as starting cells or surfaces. For this analysis, only the energy distributions were user specified (see Tables 5-1, 5-3, and 5-4).

Radiative particle emission probabilities and spectra or energy group distributions are required by MCNP and other particle transport programs for radiation field modeling. The energy groups have been chosen based on radionuclides with significant particle emissions (e.g., gammas, betas) to ensure that the probability information is as evenly distributed as possible. This method

seeks to minimize over- or underestimation of the specific contribution from any of the energy groups.

Once radionuclide radiation probability data are segregated by energy, the probabilities within each group are summed to yield the overall group probability for a particular radionuclide. After the energy groups are summed for all radionuclides, the group probability data are summarized. The input data for MCNP are activity-weighted probabilities, representing the product of the radionuclide group probability and the radionuclide activity. Once the activity-weighted probabilities have been determined, each group is summed to yield the total group probability. These group probabilities are then normalized by finding the total probability for all groups and dividing each groups' probability by the total.

5.4.2 Flux-to-Dose-Rate Conversion

Photon doses were directly calculated using the energy deposition/photon data from MCNP (Breisemeister 1997). Neutron doses were calculated for select problems identified later and then the neutron-to-photon dose ratio was used to estimate the other neutron doses.

MCNP (Breisemeister 1997) was used to calculate the average energy deposition per photon in a 1-cm-thick disk or ring of the tissue-equivalent material defined by the ICRU (ICRU 1993), in the next 14-cm-thick disk or ring, and then in an additional 15-cm-thick disk or ring (see Figure 5-3 for the model configuration of the disks and rings). With the known photon flux (i.e., emission rate), the energy deposition KERMA (i.e., first collision dose) per unit time was calculated. This was converted to dose using the definition of a rad, which is 100 ergs/g of energy deposition in tissue. The detector thickness was chosen to provide a worst case and a range that covered the ICRU 15-cm radius sphere (i.e., the basis for deep dose) (ICRU 1993). The average energy per gram deposited was calculated in the first centimeter of tissue, a 15-cm thick layer of tissue, and a 30-cm thick layer of tissue. This provided an accurate, direct method of assessing dose and is consistent with the International Commission on Radiological Protection approach for assessing dose used by the U.S. Environmental Protection Agency, U.S. Department of Energy (DOE), and U.S. Nuclear Regulatory Commission (NRC).

The photon flux was calculated separately for each source term (i.e., the first 10 cm, the balance of the first fuel layer, and the remaining four layers of fuel; see Figure 5-1) based on the photon yield of the uranium and fission products present within the matrix. The flux for each tier's source term was then multiplied by the energy deposited per photon, the doses consolidated, and the units corrected.

The MCNP tally used to assess dose actually calculates KERMA (i.e., first collision dose). This assumes kinetic energy transferred to charged particles is locally deposited. Figure 5-7 illustrates the relationship between KERMA and exposure (Olsher and Seagraves 1999). Dose also was calculated based on flux using the conversion factors from Title 10, *Code of Federal Regulations*, Section 835, "Occupational Radiation Protection" (10 CFR 835), for neutrons and from ANSI/ANS-6.1-1991, *Neutron and Gamma-Ray Fluence-to-Dose Factors*

(ANSI/ANS 1991), for photons. The calculation based on flux results in an increase in dose by a factor of 2.9 ± 0.4 for photons and 2.0 ± 0.3 for neutrons. The MCNP manual (Breismeister 1997) indicates that the ANSI/ANS (1991) dose factors for calculating photon dose from flux result in an increase in dose by a factor of 2.3 compared with International Commission on Radiological Protection results, which appears consistent with the calculated values.

5.4.3 Dose Rates

The results of the dose rate calculations are summarized in Table 5-7. These results were calculated using the matrix as discussed above. Data are given for disk/ring detectors 1 cm, 14 cm, and 15 cm thick (see Figure 5-3); areal averages are given in the rows labeled areal average. The areal average is the area-weighted average for each of the concentric rings plus the center disk (i.e., the areal average represents the dose rate averaged over the entire surface of the shield plug). The bottom row in Table 5-7 includes the neutron dose. The neutron dose rate was based on direct calculation of neutron dose for seven locations above the MCO cask (see Table 5-8). The increase in dose rate out to about 15 cm, particularly for the thinner detectors, is due to scatter from the four penetrations through the plug. One would find a similar quad-modal variation in dose rate circumferentially due to scatter from the penetrations. The dose measured by an ionization chamber at about 2.5 cm to 5.1 cm above the MCO top would approximate the dose specified for the 1-cm-thick disk and could be considered a reasonable estimate of the instrument contact dose rate. The doses for the 14-cm- and 15-cm-thick disks would provide the probable range of actual dose to an individual consistent with the ICRU model for deep dose. The doses in Table 5-7 are for peak loading (i.e., assuming full fuel loading of the MCO). The doses for the average fuel load would be about 50% of these values. This reduction value is based on a ratio of the activation product activity for the shielding design basis fuel inventory (Table 3.8 of HNF-SD-SNF-TI-009 [Praga 1998a]) and the average basin fuel inventory (Table 3.5 of HNF-SD-SNF-TI-009 [Praga 1998a]) on a per MTU basis.

The areal average doses summarized in Table 5-7 provide a reasonable estimate of the doses to an individual near the surface of the MCO shield plug. The 1-cm detector would provide an upper bound of the dose rate, but a more reasonable range of dose rate to the worker would be the dose rates for the 14-cm- and 15-cm-thick detectors. Thus the dose rate for a worker will be less than 8 mrem/h and should be in the range from 3 to 5 mrem/h. The various radii of detector rings provide information on how the dose rate varies as a function of the radius from the center of the shield plug. The neutron doses were based on the data from Table 5-8.

The doses calculated for design of the MCO shield plug are not intended to address the doses of concern for operational activities, although these doses could be used for estimating the integrated doses for workers whose activities occur above the MCO shield plug when the cask lid is removed. Because the local areas on the MCO shield plug immediately above and around the four plug penetrations will have elevated dose rates, evaluations will be conducted in accordance with each facility's ALARA (as low as reasonably achievable) program and appropriate protective measures will be taken. The current design also considers other system goals such as maximizing the payload and minimizing the number of trips to minimize the impact on the public.

Table 5-7. Peak Dose Rates Above the Multi-Canister Overpack Shield Plug.

Dimensions of ring detector		1-cm thick detector located just above the shield plug		14-cm thick detector located between 1 and 15 cm above the shield plug		15-cm thick detector located between 15 and 30 cm above the shield plug	
		Photon dose rate (mrem/h)		Photon dose rate (mrem/h)		Photon dose rate (mrem/h)	
Inner radius (cm)	Outer radius (cm)	Value	Uncertainty (2 sigma)	Value	Uncertainty (2 sigma)	Value	Uncertainty (2 sigma)
0	1	2.9	4.3	10.5	13.4	3.9	4.2
1	5	7.3	9.0	9.4	5.3	5.5	3.5
5	10	26.8	11.3	19.3	7.6	7.0	3.4
10	15	41.7	10.9	22.2	6.1	6.8	2.5
15	20	11.9	5.4	12.7	3.7	5.6	1.7
20	25	3.1	2.5	4.6	1.4	4.0	1.3
25	30	5.3	3.5	4.3	1.6	3.0	1.0
30	32.5	7.1	5.8	6.1	3.0	3.1	1.5
Photon areal average		9.9	2.3	8.0	1.2	4.0	0.7
		Neutron dose rate (mrem/h)		Neutron dose rate (mrem/h)		Neutron dose rate (mrem/h)	
		Value	Uncertainty (2 sigma)	Value	Uncertainty (2 sigma)	Value	Uncertainty (2 sigma)
Spontaneous fission neutrons		2.2E-01	6.8E-03	4.9E-02	1.3E-03	1.4E-02	6.6E-04
Alpha-n neutrons		1.8E-01	8.9E-03	3.9E-02	1.8E-03	1.2E-02	1.1E-03
Total neutrons		4.0E-01	7.4E-02	8.9E-02	3.6E-02	2.5E-02	2.7E-02
Photon areal average plus neutron dose rates		10.3	2.3	8.1	1.2	4.1	0.7

Note: For average fuel loading, dose rates are about 50% of peak values.

Table 5-8. Localized Dose Rates for Detectors of Various Thicknesses of the International Commission on Radiation and Measurements-Equivalent Tissue. (2 sheets)

Detector*	Geometry	Photon dose rate (mrem/hr)		Spontaneous neutron dose rate (mrem/hr)		Alpha-n neutron dose rate (mrem/hr)	
		Value	Uncertainty (2 sigma)	Value	Uncertainty (2 sigma)	Value	Uncertainty (2 sigma)
Localized dose for a 1-cm-thick detector located just above the surface of the shield plug							
40	Center	7.8E+00	3.2E+00	2.3E-01	9.3E-03	1.9E-01	9.8E-03
41'	Over rupture disk, filtered process port, and sensor port	2.2E+02	5.5E+01	2.7E-01	1.3E-02	2.2E-01	2.0E-02
42'	Off-center of the rupture disk, filtered process port, and sensor port	1.4E+02	5.3E+01	2.5E-01	1.2E-02	2.1E-01	2.0E-02
43'	Beyond rupture disk, filtered process port, and sensor port over internal penetration	1.9E+01	1.2E+01	1.9E-01	9.9E-03	1.5E-01	1.7E-02
41	Over long tube	1.9E+02	5.6E+01	2.8E-01	1.7E-02	2.2E-01	1.1E-02
42	Off-center of the long tube	9.4E+01	5.8E+01	2.6E-01	1.6E-02	2.0E-01	1.0E-02
43	Beyond long tube over internal penetration	9.0E+00	5.0E+00	2.0E-01	1.3E-02	1.7E-01	1.0E-02
Localized dose rates for a 12-cm-thick detector located between 1 cm and 16 cm above the shield plug							
50	Center	2.8E+01	1.5E+01	5.6E-02	2.9E-03	4.4E-02	3.1E-03
51'	Over rupture disk, filtered process port, and sensor port	8.3E+01	2.3E+01	5.1E-02	2.4E-03	4.2E-02	3.8E-03
52'	Off-center of the rupture disk, filtered process port, and sensor port	6.9E+01	1.8E+01	5.8E-02	3.0E-03	4.4E-02	4.0E-03
53'	Beyond rupture disk, filtered process port, and sensor port over internal penetration	3.4E+01	1.9E+01	4.3E-02	2.3E-03	3.3E-02	3.4E-03
51	Over long tube	7.3E+01	2.3E+01	5.1E-02	3.0E-03	4.4E-02	2.4E-03
52	Off-center of the long tube	4.0E+01	1.8E+01	5.8E-02	3.8E-03	4.4E-02	2.3E-03
53	Beyond long tube over internal penetration	2.3E+01	1.7E+01	4.0E-02	2.6E-03	3.3E-02	1.9E-03

Table 5-8. Localized Dose Rates for Detectors of Various Thicknesses of the International Commission on Radiation and Measurements-Equivalent Tissue. (2 sheets)

Detector*	Geometry	Photon dose rate (mrem/hr)		Spontaneous neutron dose rate (mrem/hr)		Alpha-n neutron dose rate (mrem/hr)	
		Value	Uncertainty (2 sigma)	Value	Uncertainty (2 sigma)	Value	Uncertainty (2 sigma)
Localized dose for a 1.5-cm-thick detector located between 16 cm and 31 cm above the shield plug							
60	Center	1.9E+01	8.1E+00	1.4E-02	1.0E-03	1.2E-02	1.3E-03
61'	Over rupture disk, filtered process port, and sensor port	1.5E+01	5.1E+00	1.2E-02	1.2E-03	1.1E-02	2.3E-03
62'	Off-center of the rupture disk, filtered process port, and sensor port	1.2E+01	4.6E+00	1.7E-02	1.5E-03	1.3E-02	2.4E-03
63'	Beyond rupture disk, filtered process port, & sensor port over internal penetration	1.8E+01	1.3E+01	1.1E-02	1.2E-03	9.2E-03	2.1E-03
61	Over long tube	1.1E+01	5.1E+00	1.2E-02	1.5E-03	9.3E-03	9.5E-04
62	Off-center of the long tube	1.6E+01	1.7E+01	1.5E-02	1.6E-03	1.4E-02	1.4E-03
63	Beyond long tube over internal penetration	1.2E+01	1.2E+01	1.0E-02	1.4E-03	9.4E-03	1.1E-03

*Detector numbers refer to locations shown in Figures 5-11 through 5-13.

In addition to the areal average doses, doses over specific localized areas of interest also were calculated for neutrons and photons (see Table 5-8 and Figures 5-8 through 5-13). These calculations were also the basis of the neutron-to-photon dose ratio used to assess total areal dose. The localized areas of interest are the center of the cask, directly above the tops of the four major penetrations, above the edge of the cask, and half way between the detectors at the penetrations and edge of the cask. The detector sets are 5-cm radius cylinders that are 1 cm and 15 cm-thick, except for one of the edge detectors, which is a 15-cm radius ICRU sphere. The detector doses are calculated in the same way as those for the ring detector. The ICRU tissue detector allows an evaluation of an individual deep dose because the detector is not sufficiently large to ensure that electronic equilibrium would have occurred, but dose is averaged over a large area. Note that the neutron dose rates will provide a reasonable upper bound of the general neutron dose rate above the shield plug. As shown, the neutron dose rate will not be a significant source of worker exposure compared to the photon dose rate.

The MCO handling operations also will involve personnel working in the area when the cask is filled with water. When the cask and MCO are filled with water the photon dose is reduced by a factor of 2. For the neutron dose, calculations showed that the increased absorption of the thermalized neutrons by the detector compensates approximately for the loss of neutron flux due to the presence of water.

The MCO also was analyzed with a basket of fuel scrap in the top layer of the MCO. In this case, both the Mark IV and Mark IA fuel sources were analyzed to ensure the results were conservative. The basket was assumed to contain a homogenized fuel-steel-air mixture; the mass of fuel was assumed to be equivalent to the basket criticality limits for the two fuel types. The doses are summarized in Table 5-9. The increased distance to the MCO shield plug from the Mark IA fuel basket was not included in the analyses. Reduction in distance should have no significant effect because the photon beam is well collimated.

The calculated dose rates summarized in Table 5-9 for the scrap basket provide an extremely conservative upper bound. The calculated dose rate values are based on the presence of the maximum source inventory in the basket with the minimum shielding density (i.e., the source material is assumed to be uniformly spread throughout the entire basket). The actual dose rates measured during operations are expected to be below the dose rates associated with a layer of fuel as described in Table 5-7 because scraps would have a significantly higher shielding density than the layers of fuel elements and normally will be well below the fuel mass criticality limit used in the source assumption.

Note that a comparison of the photon dose rates calculated for the Mark IV scrap basket (Table 5-9) versus fuel basket (Table 5-7) shows that self-shielding of photons by the fuel has a substantial impact. Indeed, homogenizing the basket of scrap in the MCNP model results in an extremely conservative dose value because material density is minimized, and hence self-shielding, for the mass of material. The scrap basket also was assumed to be completely filled with fuel and loaded at the criticality mass limit. The calculated areal average dose rates are a factor of 2 higher for the scrap basket versus the fuel basket for the 1-cm-thick detector and factors of approximately 1.6 and 0.8 for the 15-cm and 30-cm detectors, respectively.

Table 5-9. Multi-Canister Overpack Scrap Basket Dose Calculation.

Dimensions of ring detector		1-cm thick detector located just above the shield plug		14-cm thick detector located between 1 and 15 cm above the shield plug		15-cm thick detector located between 15 and 30 cm above the shield plug	
		Photon dose rate (mrem/h)		Photon dose rate (mrem/h)		Photon dose rate (mrem/h)	
Inner radius (cm)	Outer radius (cm)	Value	Uncertainty (2 sigma)	Value	Uncertainty (2 sigma)	Value	Uncertainty (2 sigma)
Mark IV fuel							
0	1	2.0E+01	1.3E+01	4.8E+00	1.3E+00	1.08E+00	2.5E-01
1	5	4.5E+00	8.1E-01	4.9E+00	9.7E-01	2.5E+00	6.6E-01
5	10	1.4E+01	3.0E+00	9.2E+00	1.7E+00	2.6E+00	6.0E-01
10	15	2.2E+01	3.0E+00	1.4E+01	2.2E+00	4.4E+00	9.1E-01
15	20	1.0E+01	1.9E+00	6.9E+00	9.8E-01	3.2E+00	6.8E-01
20	25	3.8E+00	7.7E-01	3.9E+00	7.8E-01	2.4E+00	5.9E-01
25	30	6.0E+00	1.5E+00	7.2E+00	3.2E+00	2.6E+00	1.4E+00
30	32.5	3.0E+01	1.8E+01	1.1E+01	6.6E+00	1.6E+00	5.5E-01
Photon areal average		1.4E+01	5.0E+00	8.1E+00	2.0E+00	2.5E+00	4.3E-01
Photon areal average plus neutron dose rates*		1.6E+01	5.5E+00	8.9E+00	2.2E+00	2.7E+00	4.8E-01
Mark IA fuel							
0	1	1.3E+01	7.9E+00	2.5E+00	1.5E+00	1.2E+00	1.2E+00
1	5	3.9E+00	5.6E-01	3.2E+00	1.3E+00	1.1E+00	5.1E-01
5	10	1.6E+01	3.4E+00	4.5E+00	1.3E+00	1.2E+00	4.2E-01
10	15	2.4E+01	3.0E+00	7.1E+00	1.7E+00	2.0E+00	7.5E-01
15	20	8.4E+00	1.2E+00	3.3E+00	8.9E-01	1.4E+00	4.7E-01
20	25	3.4E+00	5.5E-01	2.0E+00	5.6E-01	9.8E-01	3.9E-01
25	30	5.0E+00	1.0E+00	2.8E+00	2.0E+00	1.0E+00	8.6E-01
30	32.5	2.1E+01	1.1E+01	3.7E+00	4.1E+00	6.5E-01	3.5E-01
Photon areal average		1.1E+01	3.1E+00	3.4E+00	1.3E+00	1.1E+00	2.8E-01
Photon areal average plus neutron dose rates*		1.3E+01	3.4E+00	3.7E+00	1.4E+00	1.2E+00	3.0E-01

*The neutron dose rates were conservatively assumed to be 10% of the photon dose rates based on Table 5-8 calculational results.

A recent change in the MCO guard plate design subsequent to the dose rate modeling resulted in an increase in the number, size, and position of the holes in this plate. This change was evaluated to determine its impact on the calculated dose rates. Based on this evaluation, the revised plate design resulted in an increase of approximately 2% in the photon dose rates. No significant change in the neutron dose rates are expected because the neutron dose rates are not significant compared with the photon dose rates and are not significantly impacted by this design change.

5.4.4 Uncertainties

The presence of a steel plug over the central cylindrical tube (i.e., a steel plug within the MCO that is different from the MCO shield plug) was not included in the original model for the photon calculations as part of the ongoing shielding analysis. However, the reduction in doses from this plug was estimated using MCNP (Breisemeister 1997). This dose will range from 100% to 45% of the original values depending on the location. In the central area, where most of the areal dose is generated, the dose will range from about 100% to 65%. Overall, the plug introduces less than a 10% measure of conservatism for the doses.

The MCNP software automatically provides an assessment of the statistical relative errors or convergences (i.e., one sigma) of the tally values of flux or energy deposition (Breisemeister 1997). By the Central Limit Theorem, as the number of particles approaches infinity, there is a 68% chance that the true result will be in the range of the calculated value $\times (1 \pm R)$, where R is the estimated relative error or convergence. This confidence statement refers to the precision of the MCNP calculation itself and not to the accuracy of the result compared to the true physical value. A statement regarding the accuracy requires a detailed analysis of the uncertainties in the physical data, modeling, sampling techniques and approximations used in the calculation.

The MCNP program also allows using biasing techniques to send a predominant amount of the generated radiative particles into particular regions to help produce statistically significant results in those regions. These biasing techniques do not influence the results because the program appropriately weights these particles. However, use of these biasing techniques must be done with care after proper consideration of all of the significant tally (e.g., flux or energy deposition) contribution sources. Also, false convergences may be observed for particles that encounter significant scattering (e.g., neutrons) if insufficient scattering of the neutrons has occurred in the problem run relative to their entering the region of concern.

The MCO shield plug MCNP analyses used biasing to emphasize both the photons moving up towards the shield plug and the photons with higher energies that were more likely to penetrate the shield plug. The biasing of the photon source energy distribution was done using importance weighting as summarized in Table 5-10. This biasing technique, known as geometry splitting with Russian roulette, was used in the MCNP runs to shorten run times. This technique of variance reduction relies on the use of importances to get MCNP to either split or eliminate particles. As a particle enters a cell with an importance of X , the ratio of entering versus exiting

importance is calculated: X/Y (Y being the exiting cell importance). If the ratio is greater than one, the entering particle is split into the ratio number of particles (for nonintegers, this is done probabilistically). By doing this, the particle population increases as the particles move toward the area of interest (i.e., detector locations). Conversely, if the ratio is less than one, the population is decreased based on the ratio value. Proper use of this technique can increase the convergence of a detector set while decreasing the overall run-time.

Table 5-10. Photon Energy Biasing.

Energy (MeV)	Importance
<0.3	0.1
0.4 and 0.5	0.5
0.6	1
>0.6	5

Areas where the flux would diverge from the direction of the MCO shield plug were assigned importances that were a factor of 2 to four less than areas associated with the general movement of the flux towards the MCO shield plug. Areas outside of the MCO, aside from the detector area directly above the shield plug, were assigned an importance of zero. Note that while the MCNP program uses the importances to select the biasing for energy or direction of a particle, it will renormalize the results upon completion of the analysis.

The net variation in importance between the bottom of the cask and the top detector located above the MCO shield plug exceeded a factor of 100. Generally the difference in importance between two adjacent cells was limited to two to four. In some cases this value reached six for lower portions of the cask, which would not contribute significantly to the dose above the MCO shield plug. To assess the impact of this enhanced variance reduction in these localized area in the lower MCO, a run was made with the importances set to one. It was found that the results eventually converged to the same value. This variation approach was not expected to affect the results because the only doses of interest result from the flux passing up through the cask and through the shield plug.

In addition to the importance biasing, other actions were taken to bias the results toward a conservative answer. First, the MCO cask wall thickness was not known at the time of the analysis, so the cask wall was assumed to be a 20-cm-thick steel wall to maximize reflection of photons back into the upward flux towards the MCO cask. In addition, the top layer of source was broken into a 10-cm top section and the balance of the first layer of fuel elements because most of the dose will come from the top 10-cm of uranium as a result of the self-shielding of the fuel element. The results for each layer (i.e., the top 10 cm of the upper layer of fuel, the balance of the upper layer of fuel, and the balance of the fuel) were analyzed separately and the results

then multiplied by the activity present in each area to produce the final dose. Note that the lowest fuel tier layers are not significant contributors (i.e., <1%) to the total photon dose. In all cases the entire fuel matrix was present and provided shielding.

To reduce the uncertainty in the calculations, each of the various areal source locations (see Figures 5-3 and 5-1) was analyzed for the inner and outer ring of source material, and then these values were consolidated. In addition, the upper 10 cm of the first layer (see Figure 5-1) were analyzed in the greatest detail because this layer would be a major contributor to the dose above the cask based on the shielding characteristics of uranium and the behavior of photons.

When using a Monte Carlo program it is necessary to use enough particles (i.e., case studies) to adequately represent the behavior of the system (i.e., obtain convergence of the predicted results with the true results). This must always be a compromise between the time required to run the calculation and the amount of convergence obtained. For example, over 10,000,000 particles (i.e., case studies) were run for each source term analyzed. Because each fuel element or section of a fuel element (i.e., the upper 10 cm of the first fuel element) has an inner and outer portion (see Figures 5-5 and 5-6), this means that each of these portions was run for over 10,000,000 particles. Because of the limitation of time, the goal was to achieve a convergence of less than 0.2 for detectors that are significant dose contributors. The convergence values represent the estimated relative errors corresponding to one standard deviation. Other detectors would have lower convergences for a specific source. The higher values of convergences have a larger uncertainty so the doses were recalculated using values with convergence results of less than 0.3, and typically less than 0.2, for one of the analyses. The results of this recalculation were comparable, so it is reasonable to assume that the estimate is reasonable and conservative. Because the base source geometry is simple, it is reasonable to assume that the area where there would be significant dose contributions would be the area where the convergence is good, as the photon flux in this area is the highest. Note that the top layer of source (i.e., the upper 10 cm of the first fuel elements) requires 108 separate runs and each run requires about 7 hours to complete, so the 10,000,000 particles analysis results reflect a reasonable balance of accuracy and computation time.

The dose results shown in Table 5-7 for the MCO without the cap (i.e., the steel plug over the central cylindrical tube) indicate a higher degree of accuracy than the results with the cap in place. In this scenario, convergence values are all lower than 0.2 except for the locations that do not contribute significantly to dose. In areas where there are significant contributors to the dose rate, the MCNP results must meet the convergence criterion (i.e., <0.2). In areas where the dose contribution is not significant (e.g., the small areas in the center of the MCO, which have no source beneath them), greater uncertainty is allowed. This greater uncertainty is allowed because obtaining better convergence in these areas would require significant additional run times and/or special analyses to meet the convergence criterion with minimal benefit to the assessment of the dose rates associated with the MCO shield plug. Sufficient cases (i.e., at least 10,000,000 particles) were run to ensure that the major dose rate contributors were not missed and to provide a reasonable measure of these dose rates. The dose rates for the regions of less dose rate significance were compared with adjacent regions to ensure that the results were consistent even though the uncertainty (i.e., convergence) may have been higher than that

required for the significant contribution regions. The intent was to balance cost, run time, and dose rate accuracy so that the dose rate information of interest was accurately assessed with a reasonable expenditure of resources.

The uncertainty data provided in this report are based on the computational uncertainties (i.e., convergence values) associated with the Monte Carlo analysis. When values are calculated based on these uncertainties, the uncertainty for the new value is calculated based on a combination of the following propagation of error equations from *Data Reduction and Error Analysis for the Physical Sciences* (Bevington 1969).

$$\text{Addition/subtraction: } (A \pm a) = (C \pm c) + (D \pm d) + (E \pm e)$$

where

$$a = (c^2 + d^2 + e^2)^{1/2}$$

$$\text{Multiplication/division: } (A \pm a) = (C \pm c) * (D \pm d) * (E \pm e)$$

where

$$a = [(c/C)^2 + (d/D)^2 + (e/E)^2]^{1/2}$$

$$\text{Multiplication/division: } (A \pm a) = (C \pm c) * k$$

where

$$a = c * k.$$

The statistical uncertainties based on the convergence data are provided for the two-sigma confidence interval for the data. Uncertainties are propagated consistent with standard statistical methodology (Bevington 1969).

Figure 5-1. Model of Multi-Canister Overpack and Fuel Used for Calculations.

Note: All stainless steel material regions were modeled as homogeneous regions. Different cross hatching was used merely to visually differentiate the various regions and not to indicate different material compositions. Fuel regions were modeled as shown in Figures 5-4 and 5-5.

Figure 5-2. Details of the Multi-Canister Overpack Shield Plug Monolith.

NOTE: All dimensional values are in inches.

Figure 5-3. Modeling Data for Multi-Canister Overpack Shield Plug.

Note: All cross-hatched regions represent separate stainless steel regions.

Figure 5-4. Fuel Geometry, Mark IV Fuel Element Assembly.

Figure 5-5. Fuel Array Geometry, Mark IV Fuel Elements.

Figure 5-6. Details of General Monte Carlo N-Particle Transport Code Model for the Multi-Canister Overpack.

Figure 5-7. Relationship Between KERMA and Exposure.

Figure 5-8. Areal Average Photon Dose Rate Data for
One-Centimeter-Thick Detector.

Figure 5-9. Areal Average Photon Dose Rate Data for
Fourteen-Centimeter-Thick Detector.

Figure 5-10. Areal Average Photon Dose Rate Data for
Fifteen-Centimeter-Thick Detector.

Figure 5-11. Localized Photon and Neutron Dose Rates to a One-Centimeter-Thick Detector Located Just Above the Surface of the Shield Plug.

p = photon dose rate in mrem/h.
n = neutron dose rate in mrem/h.
Small circles show the detector numbers.

Figure 5-12. Localized Photon and Neutron Dose Rates to a Fifteen-Centimeter-Thick Detector Located Between One and Sixteen Centimeters Above the Shield Plug.

p = photon dose rate in mrem/h.
n = neutron dose rate in mrem/h.
Small circles show the detector numbers.

Figure 5-13. Localized Photon and Neutron Dose Rates to a Fifteen-Centimeter-Thick Detector Located Between Sixteen and Thirty-One Centimeters Above the Shield Plug.

p = photon dose rate in mrem/h.
n = neutron dose rate in mrem/h.
Small circles show the detector numbers.

Figure 5-14. Spontaneous Fission Spectra in MCNP.

6.0 CRITICALITY EVALUATION

This chapter identifies, describes, discusses, and analyzes the criticality safety physics used for design of the multi-canister overpack (MCO) and its components and systems that are important to criticality safety. An MCO may be loaded in one of three normal configurations:

- All fuel baskets
- One scrap basket and four or five fuel baskets
- Two scrap baskets and three or four fuel baskets.

The design of the MCO has evolved since the initial criticality analyses were performed. Some of the initial analyses have been reexamined and new analyses have been performed. The discussion in this chapter reflects the various stages of the MCO design's evolution. The criticality safety evaluations described in HNF-SD-SNF-CSER-005 (Kessler 2000) are the most current available, and both this chapter and HNF-SD-SNF-CSER-005 use the MCO loading with the highest k_{eff} as the basis for acceptability.

6.1 SUMMARY

6.1.1 Discussion and Results

The MCO is required to meet criticality criteria that are consistent with commercial reactor fuel handling (Garvin 1997b). Therefore, the k_{eff} (a measure of effective neutron multiplication constant) of the MCOs during transport to and handling in the Cold Vacuum Drying Facility (CVDF) and Canister Storage Building (CSB) must be less than 0.95 under both normal conditions and credible off-normal conditions. Without internal moderation, uranium with enrichments of less than 5 wt% ^{235}U cannot go critical (LANL 1996). Analysis described in this chapter shows the k_{eff} of the contents of the cask-MCO package under normal conditions to be below this limit by a substantial margin. Dry cask-MCO packages typically have a k_{eff} less than 0.4. Cask-MCOs flooded with water and loaded with the normal loading of spent nuclear fuel and scrap have a k_{eff} less than 0.95. In summary, the MCO container, holding dry fuel, cannot be made critical under any conditions.

The major criticality concern for a loaded MCO is internal flooding. MCOs containing either Mark IV or Mark IA fuel are below a k_{eff} of 0.90 for flooded intact fuel loadings. MCOs containing scrap in the top and bottom tiers with intact fuel in the other tiers have a k_{eff} that is on the order of 0.92. If an MCO is misloaded so that two scrap baskets are adjacent, the k_{eff} increases but is still below 0.95. Under normal conditions, a Mark IV basket loaded with scrap is more limiting because of the absence of any safety-class center post that excludes scrap from the center of the fuel basket; the Mark IA scrap basket has such a post and the amount of scrap loaded in the Mark IA scrap basket is limited.

The masses used in the criticality analysis represent the optimum fuel-to-moderator ratios of fuel scrap involved in potential handling accidents in the K Basins. These cases are discussed in the fuel retrieval and MCO loading safety analysis documents, and are incorporated into the K Basin safety analysis report (SAR). Partially loaded fuel baskets are discussed in Sections 6.4.5.7 and 6.4.5.8. Also, the amount of scrap allowed to be loaded into a Mark IA scrap basket is limited to 575 kg while the Mark IV scrap basket is limited to 980 kg (Kessler and Peck 2000). Therefore, a fully loaded Mark IV MCO contains more fissile material than a fully loaded Mark IA MCO producing a larger k_{eff} . Fuel types are to be segregated by enrichment. Because Mark IV material is less reactive than Mark IA material, Mark IV assemblies or 0.95 wt% ^{235}U scrap may be loaded into baskets designed for Mark IA fuel. Mark IA and Mark IV fuel will normally be handled separately. Procedures will be written to control the retrieval of Mark IA fuel and Mark IA scrap so that bringing a canister from an incorrect location in the K Basins will require at least two unrelated errors and also will require unidentified material to be handled as Mark IA material. The 12 Mark IA fuel assemblies that will not fit in the Mark IA fuel baskets will be loaded into a Mark IV fuel basket.

This limit is fuel with an enrichment greater than 0.95 wt% shall only be loaded in Mark IA MCO baskets. Fuel scrap with an enrichment greater than 0.95 wt% shall only be loaded in Mark IA MCO scrap baskets. The only exception is noted below:

Exception: No more than 12 long-length (26 in.) Mark IA fuel assemblies in the KW Basin can be loaded in one Mark IV fuel basket.

This exception assumes the Mark IA fuel is added to the Mark IV fuel in the MCO basket.

The inside diameter of the MCO provides geometry control and in combination with additional constraints and limits, such as the center post and basket baseplate in the Mark IA baskets, control of fuel type or enrichment in MCO loading, and limiting the number and placement of scrap baskets, ensures k_{eff} is less than 0.95 for normal and credible off-normal conditions. For accidents, two independent, concurrent or sequential, and unlikely events must occur before k_{eff} will exceed 0.95. This is the double contingency principle of criticality safety.

Several barriers are used to ensure the double contingency principle is maintained. These are listed below for each facility:

- K Basins and Cold Vacuum Drying Facility
 1. Mass control in the loading of the fuel and scrap baskets and in the loading of the MCOs
 2. Geometry control in the design of the Mark IA fuel and scrap baskets and in the design of the MCOs

- **Canister Storage Building**
 1. Geometry control in the design of the MCOs and the design of the storage vaults
 2. Moderation control in the design of the CSB to prevent water from flooding either the vault or the storage tubes.

Administrative controls on basket loading (Kessler and Peck 2000) are used to control the mass in the baskets. Administrative controls on MCO loading (Kessler and Peck 2000, Pajunen 2000) are used to control both mass and basket location in the MCO. Operating procedures prevent the shipment of flooded MCOs to the CSB.

The MCOs and the center post and baseplate of the Mark IA fuel and scrap baskets are safety-class items designed to maintain the geometry of the fuel. None of the components of the Mark IV baskets are required to provide a safety function because the enrichment is only 0.95 wt% ^{235}U and the safety-class MCO maintains the fuel and scrap geometry for criticality safety. Geometry control of the CSB storage vault is ensured by the design of the storage tubes and support structure. The design of the CSB also ensures moderator cannot be introduced into the storage vault by minimizing any sources of water in the building.

Off-normal conditions, including design basis drops, misloaded MCOs, and flooding of the CSB have been evaluated and are discussed in detail in HNF-SD-SNF-CSER-005 (Kessler 2000).

A bounding scenario was hypothesized to determine the maximum k_{eff} values computed for Mark IA and Mark IV fuel in an MCO. This scenario involves a drop accident that results in the following:

- The shipping cask breaches
- The MCO breaches
- Hydrogen is released
- An ignition source is present
- Hydrogen ignites and the Hanford Fire Department responds
- The geometry is distorted
- All fuel rubblizes to optimally sized and optimally spaced pieces
- Moderator is added to the interior of the MCO during fire fighting.

Furthermore, the MCO was assumed to contain a mass distribution of unirradiated fuel that maximizes the calculated k_{eff} . Even under these hypothetical conditions, the system was still not critical. The maximum k_{eff} computed was approximately 0.97 for a Mark IA MCO and approximately 0.93 for a Mark IV MCO.

For k_{eff} of an MCO or an array of MCOs to exceed 0.95 independent of controls requires not only the failure of design features and the failure of administrative controls but also a

sequence of hypothetical conditions that must occur concurrently. These conditions are incredible, as discussed below.

- Fuel rubblization into optimally sized particles is judged to be incredible because when discharged from the N Reactor, the individual fuel elements survived vertical drops of more than 40 ft with only minor damage, and the elements also will have been processed by the primary clean machine in the K Basins without rubblizing.
- Optimization of the fuel requires that it form pieces that are approximately twice the thickness of the original fuel elements, which is judged to be incredible.
- Rubblized fuel and scrap achieving and maintaining optimum spacing is judged to be incredible because there is nothing to support the fuel particles in an optimum matrix.

In conclusion, the k_{eff} for all credible scenarios is considerably less than 0.95 (Kessler 2000).

The subcriticality criterion, used to judge the acceptability of a calculated k_{eff} for a fissile configuration, must account for the bias inherent in the code and cross sections used, any uncertainties in the physical problem being analyzed, and the uncertainties in both the bias determination (the experimental basis) and the calculational methods. This may be stated as follows:

$$k_{\text{eff}} = k_{\text{calc}} - \Delta k_{\text{bias}} + [(\sigma_b^2 + \sigma_c^2 + \sigma_d^2 + \sigma_e^2 + \sigma_T^2)]^{1/2} < 0.95$$

where

k_{eff} is the effective multiplication constant corrected for bias and uncertainties

k_{calc} is the calculated result

Δk_{bias} is the methods bias

σ_b is the bias uncertainty

σ_c is the calculational uncertainty

σ_T is the uncertainty in temperature change

σ_d is the uncertainty in dimensional tolerances

σ_e is the uncertainty in enrichment tolerances.

In general, sigmas are the standard deviation, but here they represent a conservative estimate of the value of the standard deviation.

Since the criticality safety criterion for the SNF Project requires the probability of a k_{eff} not exceeding the limit be 95%, the bias uncertainty must be multiplied by 2 (Schmittroth and Ruben 1996) and all other uncertainties multiplied by 1.645. Hence the calculated k_{eff} , after applying the bias and uncertainties, is required to be less than the limit of 0.95 as shown in the following equation:

$$k_{\text{eff}} = k_{\text{calc}} - \Delta k_{\text{bias}} + [(2\sigma_b)^2 + (1.645^2 (\sigma_c^2 + \sigma_d^2 + \sigma_e^2 + \sigma_T^2))]^{1/2} < 0.95$$

where (Kessler 2000)

$$\begin{aligned}\Delta k_{\text{bias}} &= -0.0004 \text{ (Schmittroth and Ruben 1996)} \\ \sigma_b &= 0.005 \text{ (Schmittroth and Ruben 1996)} \\ \sigma_c &\text{ varies with each calculation} \\ \sigma_d &= 0.000546 \\ \sigma_e &= 0.001840 \\ \sigma_T &= 0.000810.\end{aligned}$$

The equation above then simplifies to

$$k_{\text{eff}} = k_{\text{calc}} + 0.0004 + [(0.01)^2 + (1.645^2)(\sigma_c^2 + 0.002083^2)]^{1/2} < 0.95.$$

6.1.2 Analyses Assumptions

The analyses performed to evaluate the acceptability of the normal and accident conditions for the MCO in the CVDF and CSB were based on the assumptions listed below.

Assumption 1	The normal loading for an MCO includes two scrap baskets, one in the top and the other in the bottom. The fuel baskets are filled with intact inner and outer fuel assemblies (48 Mark IA fuel assemblies per basket or 54 Mark IV fuel assemblies per basket).
Assumption 2	Campaign plans will separate retrieval of Mark IA and Mark IV fuel; therefore the mixing of Mark IA and Mark IV baskets in an MCO was not considered in the analyses. The only exception is the 12 long-length Mark IA assemblies in K West Basin, which will be loaded into Mark IV baskets.
Assumption 3	The credible misloading event during the fuel retrieval process is to load the mass equivalent to one full canister (14 Mark IA fuel assemblies or 233 kg rubblized Mark IA fuel) into a Mark IV fuel or scrap basket. The rubblized Mark IA fuel was modeled as 155 kg Mark IA scrap and 78 kg of Mark IV scrap.
Assumption 4	Fuel scrap was modeled as optimally spaced, optimally moderated rods at the highest enrichment for the fuel type in the analysis. The loading limits for the baskets are 575 kg Mark IA scrap and 980 kg Mark IV scrap (Kessler and Peck 2000).

- Assumption 5 Because of the variety of different irradiation times for the fuel stored in the K Basins, no credit for burnup was taken in the analysis. All fuel and scrap was modeled as being unirradiated.
- Assumption 6 Single pass reactor (SPR) fuel and N Reactor fuel will not be mixed in an MCO.
- Assumption 7 The enrichment of the 47 unidentified SPR fuel elements was modeled as 2.1 wt% ^{235}U , the highest enrichment used for experimental purposes. None of this fuel is expected to be enriched to greater than 1.25 wt% ^{235}U .
- Assumption 8 The credible MCO misloading event is one extra scrap basket, for a total of three scrap baskets in an MCO.
- Assumption 9 After loading, MCOs are shipped from the K Basins to the CVDF with the cask-MCO annulus flooded.
- Assumption 10 The only facilities where an MCO could be dropped are the K Basins and the CSB.
- Assumption 11 Following a drop, the fuel will rubblize to optimally sized pieces at an optimum packing fraction.

6.2 SPENT FUEL LOADING

6.2.1 N Reactor Fuel Description

The fuel dimensions pertinent to the criticality analysis and to criticality prevention are given in Table 6-1. This table lists the enrichments for the fresh, or green (unirradiated), fuel before it was loaded in the N Reactor. Approximately 0.072% of the fuel stored in the K Basins is fresh.

Analyses on the effects of burnup and fission product decay show that the fresh N Reactor fuel is more reactive (higher infinite criticality factor k_{∞}) than the spent fuel in spite of the presence of plutonium products in the spent fuel (Kessler 2000). Reduced uranium enrichment and the presence of fission products compensate for any increase in k_{∞} because of the plutonium content in the spent fuel. Analysis of the effect on k_{∞} of decay of fission products over a long period of time (e.g., 100 years or more) is described in HNF-SD-SNF-CSER-005 (Kessler 2000) and provides justification for use of the fresh fuel characteristics in the analyses presented in this chapter. As such, all the criticality analyses discussed in this chapter are conservatively performed for the fresh N Reactor fuel.

Table 6-1. Description of N Reactor Fresh Fuel Elements.

	Mark IV	Mark IA
Pre-irradiation enrichment of ^{235}U	0.947% enriched	1.25-0.947% enriched
Type-length code ^{1a}	E S A C	E ^b M T F
Length (cm)	66.3 62.5 58.9 44.2	66.3 53.1 49.8 37.8
Element diameter (cm)		
1. Outer of outer	6.15	6.10
2. Inner of outer	4.32	4.50
3. Outer of inner	3.25	3.18
4. Inner of inner	1.22	1.11
Cladding mass (kg)		
1. Outer element	1.09 1.04 0.99 0.79	1.07 0.88 0.83 0.66
2. Inner element	0.55 0.52 0.50 0.40	0.66 0.54 0.51 0.04
Mass of uranium in outer (kg)		
1. (0.947% ^{235}U)	16.0 15.0 14.1 10.5	
2. (1.25% ^{235}U)		13.8 11.1 10.4 7.85
Mass of uranium in inner (kg) 0.947% ^{235}U	7.48 7.03 6.62 4.94	6.84 5.49 5.12 3.90
Weighted average of uranium in element (kg)	22.7	16.3
Ratio of Zircaloy-2 to uranium (kg/MTU)	70.0 70.8 71.6 77.1	83.8 85.5 86.3 90.4
Weighted average (kg/MTU)	70.3	85.7
% of total elements	63	37
% of length type of each fuel	78 10 7 5	0.03 ^b 87 10 3
Displacement volume (L/MTU)	67	67

^{1a}Letter code differentiates the different lengths of the Mark IV or Mark IA fuel elements (i.e., a type "E" element is 66.3 cm [26.1 in.] long).

^bThere are twelve 66.3-cm-long Mark IA assemblies in the K West Basin.

MTU = metric tons of uranium.

As shown in Table 6-1, there are two types of fuel assemblies, Mark IA and Mark IV, with different lengths and uranium enrichments. Each fuel assembly consists of two coaxial, cylindrical uranium metal elements separated by spacers and clad with Zircaloy (i.e., Zircaloy-clad tube-in-tube metallic uranium geometry). The Mark IA fuel assembly has two different uranium enrichments: the inner element with an enrichment of 0.947% (0.95%) ^{235}U and the outer element with an enrichment of 1.25% ^{235}U . There are 12 Mark IA fuel assemblies and 2 Mark IA inner assemblies in the basin that are 26.1 in. long. The weighted average enrichment of a Mark IA assembly is 1.15% ^{235}U . The Mark IV assembly has only one enrichment, 0.95% ^{235}U . Some Mark IB, IC, and IVB fuel is stored in the basins. Both the Mark IB and IVB fuel contain natural uranium, and the Mark IC fuel is enriched to 0.95 wt% ^{235}U . These assemblies were considered to be Mark IV fuel for the criticality analyses. Mark IVC fuel was also stored at the K Basins but accountability records show that none of this fuel still is stored in the K Basins. The Mark IA fuel has a higher k_{inf} and, correspondingly, has more potential for criticality than the Mark IV fuel in an optimum lattice arrangement.

Most of the Mark IA fuel assemblies and scrap are stored in the K West Basin; a small amount of Mark IA fuel and scrap is stored in the K East Basin. Mark IV fuel assemblies and scrap are in both K West and K East Basins. Because of deterioration of the fuel assemblies, a significant fraction of the fuel is scrap. Removing intact fuel assemblies from existing canisters, cleaning them, and repacking them will generate additional scrap.

Operations procedures at the K Basins will keep fuel segregated by enrichment and load it in the proper baskets. This will prevent inadvertent loading of Mark IA fuel or scrap in baskets designed for Mark IV fuel or scrap. Scrap and fuel that cannot be identified will be handled as Mark IA material. Contingency analyses of misloading Mark IA fuel and scrap in Mark IV baskets have been performed (Kessler 2000) and are reported in this chapter.

Both K West and K East Basins contain SPR fuel, which is not shown in Table 6-1. The SPR fuel was fabricated with aluminum cladding. Enrichments varied from highly depleted to a maximum of 2.1 wt% enriched ^{235}U (Praga 1998a). No SPR fuel with enrichments greater than 1.25 wt% is believed to be stored in the K Basins. However, there are 47 pieces of unidentified SPR fuel in the K West Basin that were assumed to be 2.1 wt% ^{235}U enriched for the criticality calculation described in Section 6.4.2.1 and spr.1 of Table 6-5. SPR fuel constitutes 3.88 metric tons of uranium (MTU), which is about 0.3% of the total fuel inventory, approximately the equivalent of one MCO load.

SPR fuel will be loaded into Mark IA fuel and scrap baskets, which will be placed in a single MCO. The fuel baskets will be modified to contain an insert to permit the SPR fuel to be stacked 2 to 3 elements high to ensure efficient packing densities and permit all the fuel to fit into a single MCO. Because the design of the insert has not been finalized, the criticality analyses for the SPR fuel were performed by modeling the fuel as aluminum clad cylinders at optimum spacing and moderation.

6.2.2 Multi-Canister Overpack Fuel Basket Description

The design parameters for the fuel baskets are shown schematically in Figures 6-1 and 6-2 for fuel assemblies and for scrap, respectively. Both the Mark IA fuel baskets and scrap baskets contain a center post for criticality control. The post has a 6.625-in. outer diameter and a 1.75-in. inner diameter. Mark IV baskets have a small central pipe to house the long process tube used for draining at the CVDF. The scrap baskets contain two sections: an inner section for small "fines" of scrap around the central pipe, and an outer region for the larger pieces. For the Mark IA scrap basket, the inner region surrounds the central pipe.

To improve the heat removal capability of the scrap baskets, the sides of the baskets are made of copper. In addition, the outer section of each basket is divided into six subsections by copper plates. These plates serve to break up the optimal geometry used in the criticality analyses reported in this section. Their effect on k_{eff} is discussed in Section 6.3. Sketches of the MCO and the MCO storage tube are shown in Figure 6-3. These figures show the acceptable arrangements and the dimensions important to the computer model but do not show the detailed design of the baskets and MCOs.

Initial criticality analyses indicated that the Mark IA fuel, and scrap, would not meet the subcriticality criterion if the geometry and moderation were optimized. Therefore, the Mark IA fuel and scrap baskets were specially designed with a post in the center that has an outer diameter of 6.6 in. as shown in Figures 6-1 and 6-2. This post prevents fuel from accumulating in the center of the basket and reduces MCO k_{eff} to acceptable levels. Radial movement of the center post in the Mark IA baskets is limited to 2 in. for drops at the K Basins (Goldmann 2000b). There is an increase in k_{eff} due to this displacement, but k_{eff} remains below 0.95 (Kessler 2000). The baseplate of all fuel and baskets are 1.25 in. thick (Goldmann 2000a).

The Mark IV fuel baskets are designed to contain a total of 54 of the longest fuel assemblies of Mark IV design (Goldmann 2000b). Since there are only a few Mark IA assemblies longer than 20.9 in., the Mark IA basket was designed to contain 48 assemblies of a maximum length of 20.9 in. The 12 long Mark IA assemblies will be loaded into a Mark IV fuel basket and MCO. Each MCO is designed to hold five baskets containing Mark IV fuel assemblies and scrap, or six baskets containing Mark IA fuel assemblies and scrap. As shown in Table 6-1, the Mark IA fuel assemblies are shorter than the Mark IV assemblies, except for Mark IV fuel type C. This reduces the amount of fuel per Mark IA basket while permitting six baskets of Mark IA fuel per MCO. The fuel elements, when loaded in the baskets, form a tight lattice with minimal separation between neighboring fuel elements. The lattice has a hexagonal center-to-center pitch of about 2.8 in. compared with outside diameters of about 2.4 in. for the fuel assemblies.

Broken pieces of the fuel assemblies (Kessler and Peck 2000) may be treated as scrap and loaded into scrap baskets.

6.2.3 Multi-Canister Overpack Loading

The MCO shipping casks are designed to contain a single MCO and to surround the MCO with an annulus used for managing internal temperature after loading at the K Basins and during draining and drying operations at CVDF. Each MCO is loaded with five Mark IV baskets or six Mark IA baskets. The MCO loading system is designed to monitor the height of the baskets as they are lowered into the MCO and to ensure that Mark IV and Mark IA baskets are not mixed in the same MCO. Present operational plans are to load one scrap basket into an MCO. Because the amount of scrap that will be generated by operation of the fuel retrieval system is not known, the criticality analysis has been based on loading a maximum of two scrap baskets in an MCO. SPR fuel will be handled in accordance with limit 1 of HNF-SD-SNF-CSER-010, *Criticality Safety evaluation Report for the K Basin Fuel Retrieval Subproject* (Kessler and Peck 2000), as either Mark IV or Mark IA scrap depending on the enrichment. This limit states that 0.95 wt% or less SPR fuel may be handled as Mark IV scrap and that unidentified fuel will be loaded as Mark IA scrap. Mixing SPR fuel and N Reactor fuel in the same MCO is not allowed.

By design, the Mark IV fuel basket may contain a maximum of 54 fuel intact assemblies and the Mark IA fuel basket may contain 48 fuel assemblies. Filled fuel baskets were used for the criticality safety evaluations unless the analyses were done specifically for partially loaded baskets.

The scrap baskets may be loaded with fuel scrap or segments of fuel assembly components, with or without cladding. Such scrap material comprises a considerable fraction of the material stored in the K Basins and additional scrap may be generated during the fuel retrieval process. Mass limits for the Mark IV and Mark IA scrap baskets are established at 980 kg and 575 kg, respectively (Kessler and Peck 2000). Most of the criticality calculations for the MCOs were performed prior to the establishment of these mass limits and assumed the scrap baskets were filled with optimized scrap. At optimum spacing, a full Mark IV scrap basket will contain approximately 980 kg of scrap, while a Mark IA scrap basket will contain approximately 800 kg of scrap. A Mark IA scrap basket containing 575 kg of optimized scrap will be approximately five-eighths full. Thus, the calculations for the Mark IA MCO are very conservative because the analysis was done for fully loaded scrap baskets. Results of calculations using two fully loaded scrap baskets bound the loadings of partially loaded scrap baskets or of zero, one, or two scrap baskets.

6.3 DESCRIPTION OF CALCULATIONAL MODEL

Different calculational models were used in the MCNP calculations referenced in this chapter. Changes were made based on evolving designs for both MCOs and baskets. The initial design was based on an MCO inner diameter of 24 in., which was later reduced to 23.5 and 23.25 in. Because the sides of the baskets are not safety-class components, they were ignored in most models. The maximum number of fuel assemblies in the fuel baskets is limited by design. Scrap basket loading limits had not been established when most of the analyses were performed and the scrap basket regions in the MCO were modeled as being filled with optimized scrap. Optimization parameters for the scrap were calculated using MCNP and are discussed in

Appendix B of HNF-SD-SNF-CSER-010. They are listed in Table 6-2. As the MCO diameter was reduced the amount of scrap in these regions was reduced, effectively lowering the k_{eff} .

Table 6-2. Scrap Optimization Parameters.

	Mark IA	Mark IV
Scrap rod diameter	0.512 in.	0.63 in.
Clad diameter	0.5596 in.	0.675 in.
Hexagonal pitch	0.946 in.	1.102 in.
Packing fraction	0.294	0.320

The original axial model does not include the shield plug in the top of the MCO but assumes a 12-in. water gap above the top basket. Including this gap and the bottom of the transport cask, the model is 154 in. high. Calculations done to verify the acceptability of the redesigned baskets included the axial detail of the shield plug above the top basket in the MCO. The shield plug contains a filter assembly to retain particulate in the MCO during draining and sampling operations. For the criticality analysis, the plug was modeled as a solid stainless steel piece approximately 9.5 in. thick with a gap for the filter assembly and a 1.5-in. thick filter guard plate above the top MCO basket. The dimensions of the baskets are such that their combined height inside the MCO is approximately the same, irrespective of the basket type. A single scrap basket in the top or bottom positions, or an MCO containing no scrap baskets, is allowed. Whole fuel assemblies are modeled with all the zirconium cladding in place. Scrap is modeled on optimally sized rods with cladding in an optimal lattice.

The baskets are modeled as shown in Figures 6-1 and 6-2. Initial models did not include the process tube in the Mark IV baskets. This tube was included in later analyses but is not required for criticality control. For the redesigned Mark IV scrap baskets, the central process tube was not included in the model. The basket diameter was adjusted to ensure the scrap filled the basket with no water gap between the upper fuel basket and the top of the scrap. Section 6.4.5.13 discusses the effect of basket width on k_{eff} . The copper in the baskets also was not modeled. Copper is a much better absorber than the water it replaces, 3.8 b versus 0.664 b, and acts as a poison, reducing k_{eff} . In the Mark IV scrap basket, not only is water replaced by the copper, but also fuel is replaced, further reducing k_{eff} . Specific basket dimensions are discussed in Section 6.2.2.

Loading, transporting, and handling operations involve single MCOs. The CVDF has four process bays, each containing an MCO. MCOs may be processed simultaneously but only one may be drained at a time. Two MCOs are stacked per tube in the CSB during long-term interim storage. Both finite and infinite arrays are used to model the MCOs in the CSB. The impact absorbers, shown in Figure 6-3, were modeled as water in the original computer models. More

recent calculations modeled the impact absorbers as half density stainless steel, the lower 36 in. thick and intermediate 24 in. thick. This change, along with using the limited scrap basket loadings, had a negligible effect on k_{eff} . The scrap baskets were originally modeled as being filled with optimized scrap. Analyses performed for the movement and storage of baskets in K Basins (Kessler and Peck 2000) imposed limits on scrap basket mass. Other models of multiple MCOs were used for specific conditions. MCOs are sealed for long-term interim storage.

MCOs are received at the CVDF with both the internal region and shipping cask annulus flooded. The residual moisture left in the MCO after drying in the CVDF was conservatively modeled at a level above its credible limit. The normal condition is the "dry" condition inside and outside the MCOs while they are in storage or being received at the CSB in the shipping cask.

The MCO storage tubes in the CSB vault are arranged in a hexagonal matrix, shown in Figure 6-4. The storage tubes form a 10 (north-south) by 22 (east-west) array with two MCOs per tube. The tubes' center-to-center distance is 4 ft, 8 in. in the east-west direction and 4 ft, 6 in. in the north-south direction. At the operating deck of the CSB, the MCOs are handled by the MCO handling machine (MHM). Figure 6-5 depicts an MCO inside the MHM, which has 10-in.-thick steel walls in a bell-shaped configuration. The operating deck's concrete floor functions as a neutron reflector and moderator at the bottom. Collocation of two or more MCOs at the operating deck is prevented by design and operating procedures. All criticality scenarios at the operating deck are analyzed for a single MCO. However, arrays of cask-MCOs have been evaluated for transportation and show a negligible effect on k_{eff} (Kessler 2000).

For ease of modeling, the CSB was initially modeled as an infinite array of MCOs. Later models used the 10 x 22 array surrounded by concrete walls. The thickness of the walls was a minimum of 1 ft to provide maximum neutron reflection (Carter et al. 1968).

For the CSB, the off-normal condition of flooding is modeled by filling the spaces between the tubes and the MCOs with varying densities of water, from very low density dispersed water (zero to 0.1 g/cm³) to a fully flooded condition (1.0 g/cm³). As the density of water is increased above 0.1 g/cm³, the tube-to-tube neutron interaction decreases and the tubes are rapidly decoupled. Based on the results of the analysis, the water-filled MCO-tube arrangement functions like an infinite lattice in which a boundary of neutron flux symmetry is defined around each tube location. Neutron leakage is zero across this boundary, or the neutrons generated in an MCO-tube cell are fully reflected back into the cell. This infinite array configuration is depicted in Figure 6-6 as model C.

The exterior vault walls are 4.5 ft thick, the partition walls between vaults are 3 ft thick, the vault basemat is 5.5 ft thick, and the operating deck is 5.0 ft thick. The vault walls, basemat, and operating deck are constructed of reinforced high-density concrete.

Figure 6-6 also depicts model D, in which an MCO is encased in concrete. Model D is used to evaluate the effects of concrete around a single MCO to simulate a configuration of operating deck concrete when an MCO is lowered into the storage tubes, or a postulated accident scenario of an MCO falling from the MHM onto the operating area.

Modeling fuel elements or fuel scrap inside the MCO and the effect of interspersed moderation are discussed in Section 6.7. Material densities and weight fractions are provided in Table 6-3 for constituent nuclides of all materials used in the calculational models for the normal and accident analyses provided in this section.

The analysis tools used in the evaluation are the WIMS-E (Gubbins et al. 1982) and GOLF (Schwinkendorf 1994) codes, used for parametric studies and the calculation of ideal geometry critical dimensions, and the MCNP code (Breisemeister 1993), used for modeling of three-dimensional geometries in detail. The WIMS-E lattice transport code (Gubbins et al. 1982) was used to generate infinite neutron multiplication factors, k_{∞} , for lattices and two-group cross sections for use by GOLF (Schwinkendorf 1994), which was used to calculate finite radial dimensions for cylinders. Section 6.6 presents the validation of the criticality computer codes used to demonstrate the acceptability of the MCO k_{eff} .

The MCNP code (Breisemeister 1993) was used to obtain the results discussed in the following sections.

6.4 CRITICALITY CALCULATION RESULTS FOR NORMAL CONDITIONS

The calculational or experimental methods and results used to determine k_{eff} for the maximum fuel loading intended to be contained in the cask are discussed in this section. Many of the results reported in Tables 6-4, 6-5, and 6-6 are included for information only. Refer to HNF-SD-SNF-CSER-005 (Kessler 2000) for the up-to-date analyses.

6.4.1 Loading with Intact Assemblies Only

6.4.1.1 Dry Multi-Canister Overpack Loadings of Intact Assemblies. Table 6-4 contains the MCNP results for casks containing MCO baskets loaded only with intact Mark IA or Mark IV fuel assemblies. The upper section of the table contains data for dry loadings; these results clearly show that criticality is not a concern for dry MCO containers. The k_{eff} for an infinite array of dry Mark IA MCOs is only 0.3105. The k_{eff} values for dry MCOs loaded with intact Mark IV fuel assemblies are about 10% higher. This is because there are more fuel assemblies per basket and the fuel assemblies are longer, thus resulting in more uranium metal.

6.4.1.2 Flooded Multi-Canister Overpack Loadings of Intact Assemblies. When the MCOs loaded with intact fuel assemblies are fully flooded, the k_{eff} is higher, but less than 0.95. The lower part of Table 6-4 gives the calculated results for cases in which the MCO is flooded inside. For a single MCO holding six baskets of intact Mark IA fuel, fully flooded with water, the k_{eff} is 0.8591. Putting a second identically loaded cask adjacent to the first cask raises the k_{eff} less than 0.4%, illustrating the significant effect of the cask shielding for diminishing interactions between the casks in an array.

Table 6-3. Material Densities and Weight Fractions Used in Calculations.

Material	No.	Density (g/cm ³)	Isotope	Wt fraction
Mark IV and Mark IA inner elements	m1	18.58	²³⁵ U ²³⁸ U	0.009471 0.990529
Zircaloy cladding	m2	6.55	Zr	1.000
Stainless steel	m3	8.03 g/cc	C ⁵⁵ Mn N Cr Ni Fe	0.0004 0.0200 0.0100 0.1900 0.0925 0.6871
Water	m4	1.000	¹ H ¹⁶ O	0.1119 0.8881
Mark IV outer elements	m6	18.58	²³⁵ U ²³⁸ U	0.009471 0.990529
Mark IA rubble	m7	18.82	²³⁵ U ²³⁸ U	0.011494 0.988506
1.25 wt% ²³⁵ U scrap, Mark IA outer elements, and SPR uranium cores	m8	18.82	²³⁵ U ²³⁸ U	0.012491 0.987509
CSB concrete	m9	2.26	H O ²³ Na Mg ²⁷ Al N ³¹ P ³² O ⁸⁴ Kr Ca Ti Fe	0.0031 0.4407 0.0182 0.0376 0.0607 0.2157 0.0009 0.0009 0.0066 0.1306 0.0049 0.0788
SPR fuel aluminum cladding		2.70	Al	1.0

CSB = Canister Storage Building.

SPR = single pass reactor.

Table 6-4. Analysis Results for Intact Fuel in Multi-Canister Overpacks in Shipping Casks when Reflected by Water.

File ID	Fuel type ^a	Payload cluster		MCO interior	Configuration analyzed	Calculation results ^b		
		Number of baskets	Assemblies per basket			k_{calc}	σ_{calc}	k_{eff}
plan_b83	Mark IA	6	48	Dry	Two side by side, touching	0.3028	0.0016	0.3131
plan_104	Mark IA	6	48	Dry	Infinite array lattice	0.3008	0.0015	0.3121
plan_b90	Mark IV	5	54	Dry	Two side by side, touching	0.3293	0.0016	0.3406
plan_103	Mark IV	5	54	Dry	Infinite array lattice	0.3341	0.0016	0.3454
plan_b81	Mark IA ^c	6	48	Flooded	Single MCO	0.8483	0.0032	0.8605
plan_b82	Mark IA	6	48	Flooded	Two side by side, touching	0.8515	0.0028	0.8634
plan_b88	Mark IV ^c	5	54	Flooded	Single MCO	0.8778	0.0028	0.8897
plan_b89	Mark IV	5	54	Flooded	Two side by side, touching	0.8794	0.0031	0.8915

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

^aMark IV type neglects the center post, which excludes center assembly in array. Mark IA has a 6.6-in. diameter steel insert, excluding assemblies from seven central positions.

^b $k_{eff} = k_{calc} + 0.0004 + ((0.01)^2 + (1.645^2)(\sigma_c^2 + 0.002083^2))^{\frac{1}{2}}$ as defined in Section 6.1. This value should be less than 0.95.

^cCask-MCO surrounded by 25.0 cm water.

MCO = multi-canister overpack.

Table 6-5. Analysis Results for Mark IA Fuel and Scrap and Single-Pass Reactor Fuel in Flooded Multi-Canister Overpacks in Casks.

File ID	Basket contents by tier number (from top)						MCO interior	Calculation results ^a		
	Tier 1	Tier 2	Tier 3	Tier 4	Tier 5	Tier 6		k_{calc}	σ_{calc}	k_{eff}
CASE1A	1.25 wt% ²³⁵ U scrap	48 Mark IA	48 Mark IA	48 Mark IA	48 Mark IA	1.25 wt% ²³⁵ U scrap	Dry	0.3808	0.0019	0.3922
CASE1	1.25 wt% ²³⁵ U scrap	48 Mark IA	48 Mark IA	48 Mark IA	48 Mark IA	1.25 wt% ²³⁵ U scrap	Flooded	0.8721	0.003	0.8842
spr.1	0.95 wt% ²³⁵ U SPR fuel	0.95 wt% ²³⁵ U SPR fuel	0.95 wt% ²³⁵ U SPR fuel	0.95 and 2.1 wt% ²³⁵ U SPR fuel	0.95 wt% ²³⁵ U SPR fuel	0.95 wt% ²³⁵ U SPR fuel	Flooded	0.8351	0.0030	0.8472

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-C-SER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

^a $k_{eff} = k_{calc} + 0.0004 + ((0.01)^2 + (1.645^2)(\sigma_c^2 + 0.002083^2))^{1/2}$ as defined in Section 6.1. This value should be less than 0.95.

^bThe 2.1 wt% ²³⁵U region was modeled as a 4 in. cylinder in the tier 4 basket around the central insert.

MCO = multi-canister overpack.
SPR = single-pass reactor.

Table 6-6. Analysis Results for Mark IV Fuel and Scrap in Flooded and Drained Multi-Canister Overpacks in Casks. (2 sheets)

File ID	Basket contents by tier number ^a					MCO interior	Calculation results ^b		
	Tier 1	Tier 2 ^c	Tier 3 ^c	Tier 4 ^c	Tier 5		k _{calc}	σ_{calc}	k _{eff}
plan_108 ^d	Mark IV inner element	Mark IV inner element	Mark IV inner element	Mark IV inner element	Mark IV inner element	Flooded	0.6762	0.0024	0.6879
CASE2A ^d	0.95 wt% ²³⁵ U scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% ²³⁵ U scrap	Dry	0.3866	0.0022	0.3982
CASE2 ^d	0.95 wt% ²³⁵ U scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% ²³⁵ U scrap	Flooded	0.8824	0.0022	0.8940
18rc.3	0.95 wt% ²³⁵ U scrap	53 Mark IV with an inner element only in location 3,6	35 Mark IV with an inner element only in location 3,6 18 Mark IA in the center	53 Mark IV inner element only in location 3,6	0.95 wt% ²³⁵ U scrap	Flooded	0.9160	0.0009	0.9271
18rc.15	0.95 wt% ²³⁵ U scrap	53 Mark IV with an inner element only in location 4,5	35 Mark IV with an inner element only in location 4,5 18 Mark IA in the center	53 Mark IV inner element only in location 4,5	0.95 wt% ²³⁵ U scrap	Flooded, annulus drained	0.9245	0.0009	0.9356
18rm.7	Scrap	36 Mark IV 18 Mark IA in the middle row	53 Mark IV with an inner element only in location 3,6	53 Mark IV with an inner element only in location 3,6	Scrap	Flooded	0.9141	0.0009	0.9252
18rm.17	Scrap	53 Mark IV with an inner element only in location 4,5	53 Mark IV with an inner element only in location 4,5	36 Mark IV 18 Mark IA in the middle row	Scrap	Flooded, drained annulus	0.9225	0.0009	0.9336

Table 6-6. Analysis Results for Mark IV Fuel and Scrap in Flooded and Drained Multi-Canister Overpacks in Casks. (2 sheets)

File ID	Basket contents by tier number ^a					MCO interior	Calculation results ^b		
	Tier 1	Tier 2 ^c	Tier 3 ^c	Tier 4 ^c	Tier 5		k _{calc}	σ _{calc}	k _{eff}
18rp.1	Scrap	53 Mark IV with an inner element only in location 4,5	35 Mark IV with an inner element only in location 4,5 18 Mark IA in outer row	53 Mark IV with an inner element only in location 4,5	Scrap	Flooded	0.9155	0.0008	0.9265
18rp.2	Scrap	53 Mark IV with an inner element only in location 4,5	35 Mark IV with an inner element only in location 4,5 18 Mark IA in outer row	53 Mark IV with an inner element only in location 4,5	Scrap	Flooded, drained annulus	0.9233	0.0009	0.9344

^aTiers number from the top to the bottom of the MCO.^b $k_{eff} = k_{calc} + 0.0004 + ((0.01)^2 + (1.645^2)(\sigma_c^2 + 0.002083^2))^{1/2}$ as defined in Section 6.1. This value should be less than 0.95.^cHighest k_{eff} base case is 53 Mark IV assemblies with an inner element only in one location in the middle row.^dThe results in this table are for information only. For current results, refer to Kessler, S. F., 2000, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*, HNF-SD-SNF-CSE-005, Rev. 5, Fluor Hanford, Incorporated, Richland, Washington.

MCO = multi-canister overpack.

Similar trends are noted for flooded casks of whole Mark IV fuel assemblies. Even though a Mark IV fuel assembly is less reactive than a Mark IA assembly, more Mark IV fuel assemblies will be loaded into each basket. This results in a value for the Mark IV k_{eff} that is 3% to 4% higher than for Mark IA fuel. The MCOs loaded with Mark IV fuel will therefore tend to be more limiting from a criticality viewpoint. When fully flooded, MCOs loaded with intact Mark IV fuel assemblies also yield a k_{eff} well below 0.95. For the side-by-side casks case, the k_{eff} is 0.8901. Analyses of arrays consisting of more than two MCOs are discussed in Section 6.5.4.

These data for the flooded casks of whole fuel assemblies prove conformance with the criticality safety requirements under normal conditions, including the transport phase.

6.4.2 Loading with Intact Assemblies and Scrap

Calculation of k_{eff} values for MCO cask loadings that include baskets of scrap or mixed scrap and intact fuel is highly dependent on the nature of the scrap. The design parameters affecting k_{eff} include average piece size and volumetric distribution, the degree of moderation, and the effective shielding of ^{238}U resonances. ANSI/ANS standards (ANSI/ANS 1983, ANSI/ANS 1984) and HNF-PRO-539, *Criticality Safety Evaluations*, require the design parameters used to evaluate k_{eff} values to be for the most reactive possible scrap configurations. Based upon the results for these hypothetical, worst-case scrap configurations, it then is inferred that any realistic scrap loadings will be less reactive than the calculational model.

6.4.2.1 Flooded Multi-Canister Overpack Loadings that Include Mark IA Scrap and Single Pass Reactor Fuel. The MCNP cases reported in Table 6-5 model casks loaded with MCOs that contain baskets of Mark IA scrap and intact Mark IA assemblies or SPR fuel.

The flooded configuration including scrap is more reactive, with k_{eff} values larger than those for loadings of intact assemblies only. Table 6-5 shows that two scrap baskets containing 1.25 wt% ^{235}U can be included in an MCO loaded with Mark IA fuel without reaching a k_{eff} of 0.95 if the baskets are placed at the top and bottom. This loading is defined as the normal loading for all criticality analyses including those for Mark IV fuel and scrap. Section 6.5.3.3 describes the contingencies of misloading scrap baskets.

CASE1A in Table 6-5 shows that for a normally loaded, dry MCO in a cask, the k_{eff} is small, less than 0.4. The dry case includes 3 kg of residual water left in the MCO after drying and is representative of the MCO with significant reflection. Its density in the MCO is 0.005 g/cm³. When this fuel loading is flooded, as in CASE1 (Figure 6-7), which represents the flooded MCO in the shipping cask with a drained annulus, the k_{eff} increases to almost 0.9. These results show that the normally loaded MCO, when isolated from other units by the cask or distance, is acceptably subcritical over the range from fully flooded to dried.

SPR fuel will be loaded into an MCO that is dedicated solely to SPR fuel; it may not be mixed with N Reactor fuel. The SPR fuel will be loaded into a stack of six Mark IA-type baskets shown in the normal spr.1 in Figure 6-8.

The geometry of the SPR fuel was modeled as aluminum-clad uranium cylinders based on representative fuel dimensions. Full spacing used in the model corresponded to optimum moderation. The depleted, natural, and 0.95 wt% ^{235}U enriched fuels were represented by 0.95 wt% ^{235}U enriched uranium. The 47 unidentified fuel elements are not expected to exceed an enrichment of 1.25 wt% ^{235}U . However, these fuel elements were modeled with uranium enrichment corresponding to 2.1 wt% ^{235}U , the highest enrichment used for special SPR fuel irradiations. The mass of the unidentified fuel was modeled at the center of the MCO loading, the center of the tier 4 basket, that corresponds to the highest neutron importance.

The k_{eff} for spr.1, shown in Table 6-5, is less than 0.95. Comparison of the k_{eff} of spr.1 with that of CASE1 illustrates that the Mark IA MCO is more limiting than the SPR MCO. Hence, the SPR MCO is bounded by the analyses of the Mark IA MCO.

6.4.2.2 Flooded Multi-Canister Overpack Loadings that Include Mark IV Scrap. The MCNP cases reported in Table 6-6 model casks loaded with MCOs that contain baskets of Mark IV scrap and intact or partial Mark IV assemblies. The results in Table 6-6 show that two scrap baskets can be included in an MCO loaded with Mark IV fuel, the same as for the Mark IA MCO. Misloaded scrap baskets are discussed in Section 6.5.3.3.

Case plan_108 (Table 6-6) models all five baskets filled with Mark IV inner elements in a close-packed hexagonal array. CASE2 models an MCO with three baskets in the center filled with whole Mark IV elements and a Mark IV scrap basket on each end. These results show that flooded intact fuel with scrap baskets is more reactive than close-packed inner elements.

CASE2A in Table 6-6 shows that for a normally loaded, dry MCO in a cask, the k_{eff} is small, less than 0.4. The dry case includes 3 kg of water as residual water left in the MCO after drying. The water density in the MCO is 0.005 g/cm³. When this fuel loading is flooded, as in CASE2 (Figure 6-7), the k_{eff} increases to almost 0.9. These results show that the normally loaded MCO, when isolated from other units by the cask or distance, is acceptably subcritical over the range from fully flooded to dried.

6.4.2.3 Handling of the Long-Length Mark IA Fuel Assemblies. The twelve 26.1-in.-long Mark IA fuel assemblies cannot fit in a Mark IA fuel basket and must be loaded into a Mark IV fuel basket. Analyses were performed to show that loading up to 18 long-length Mark IA assemblies in a Mark IV fuel basket is acceptable. Cases were analyzed to show that the Mark IA assemblies may be loaded into any location in the baskets and in any fuel basket in an MCO. When the Mark IA assemblies were modeled in either the center or outer rows, the rest of the basket contained 35 Mark IV assemblies with a single inner element in one middle row location. Because the highest k_{eff} reference loading has an inner element only loaded in the middle row, the fuel basket was modeled with 36 Mark IV assemblies in the inner two rows and the outer row and 18 Mark IA assemblies in the middle row. The MCO was loaded with two other Mark IV fuel baskets and the highest k_{eff} base case loading and two Mark IV scrap baskets. The results are summarized in Table 6-6 for the cases with the maximum k_{eff} . These results show that highest k_{eff} does not exceed 0.936 including the contingency of a flooded MCO with a drained cask annulus. A complete discussion of these results is in HNF-SD-SNF-CSER-005 (Kessler 2000).

6.4.2.4 Redesigned Fuel and Scrap Baskets. The fuel and scrap baskets have been redesigned to ease fabrication. The criticality consequences of this redesign were investigated and are reported in HNF-SD-SNF-CSER-005 (Kessler 2000). The following is a summary of the effects of the changes for the normal conditions in the MCO.

Each MCO may be loaded in one of three basic configurations: all fuel baskets; one scrap basket and the rest fuel baskets; or two scrap baskets and the rest fuel baskets. The maximum number of fuel assemblies loaded into a fuel basket is limited by design while the maximum amount of scrap loaded into each scrap basket is limited by administrative controls based on analysis of spills of scrap baskets in the K Basins (Kessler and Peck 2000). There are no minimum loading requirements for the baskets.

The beyond design basis case involving a dropped Mark IA MCO containing two full scrap baskets was analyzed to bound all possibilities of the scrap forming an unfavorable geometry in partially loaded scrap baskets. This case used the model for the drop analyses discussed in Section 6.5.2 in which four fuel baskets contain fuel rubblized to optimized scrap and two scrap baskets are filled with optimized scrap. The results in Table 6-7 show that the k_{eff} is still well below the limit of 0.95.

Table 6-7 summarizes the results of a study performed to determine the most reactive configuration of a loaded MCO (Kessler 2000). For the Mark IA MCO, the most reactive configuration is that of an MCO containing two scrap baskets with each of the four fuel baskets containing 47 assemblies. A position in row 2 of each fuel basket (location 4, 5 on the grid) is left empty. Similarly, the most reactive Mark IV MCO contains two scrap baskets and each of the three fuel baskets contains 53 fuel assemblies. Location 3, 6 in the outer row of each fuel basket, contains a single inner element. All calculations performed to verify the new design used the most reactive loadings for the MCO except for the drop analyses, which are described separately.

6.4.3 Cold Vacuum Drying Facility

The CVDF has four process bays that each house a transfer cask transporter containing a single loaded cask (see Figure 6-9). The transporter with the cask is received at the CVDF with both the MCO and cask annulus flooded. After the transporter is secured in a CVDF process bay, the lid of the cask is removed and a process hood installed. Water is circulated from a process line through the transfer cask annulus to control the internal temperature of the MCO as required for processing. The MCO is drained through the long process tube by a combination of the process water conditioning system water jet ejector, external to the MCO, and helium gas pressure applied through the internal high-efficiency particulate air (HEPA) filters.

Table 6-7. The Most Reactive Multi-Canister Overpack Configuration.

Case	Configuration	k_{calc}	σ	k_{eff}
mk1ad.5	Two full scrap baskets and four rubblized fuel baskets	0.9176	0.0007	0.9286
mk1r.0	Two scrap baskets, four fuel baskets	0.8865	0.0010	0.8976
mk1r.2	Two scrap baskets, four fuel baskets, location 4, 5 empty	0.8902	0.0010	0.9013
mk4r.0	Two scrap baskets, three fuel baskets	0.9130	0.0010	0.9241
mk4ri.4	Two scrap baskets, three fuel baskets, location 3, 6 contains inner element only	0.9154	0.0008	0.9265

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

MCNP calculation grid

	1	2	3	4	5	6	7	8	9	10	11
1											
2											
3						4ri.4					
4					1r.2						
5											
6											
7											
8											
9											
10											
11											

Shaded cells represent water in the basket (universe 2 in the MCNP code), and the unshaded cells represent fuel assemblies (universe 3 in the MCNP code).

The results of MCNP computer calculations provided in Table 6-8 show k_{eff} for normally loaded MCOs. Each MCO is isolated neutronically by the cask shielding region from other MCOs and fissionable material outside the transfer cask. The MCO and the cask's outer annulus are assumed to be flooded, which is the normal condition for casks received at the CVDF. Normal operations at the CVDF maintain the annulus flooded until the pressure test performed after MCO draining is satisfactory. Cases mk1r.2 and mk4ri.4 in Table 6-8 represent the Mark IA and Mark IV MCOs from the highest k_{eff} reference case (see Section 6.4.2.4). All the other cases in Table 6-8 represent variations of Mark IA and Mark IV fuel with the MCO either dry or wet and the cask annulus flooded. Analyses of flooded MCOs in casks with a drained annulus are described in Section 6.5.3.2. In all cases, k_{eff} is well below 0.95. Figure 6-10 shows the k_{eff} as the water is drained from the MCO.

6.4.4 Canister Storage Building

The vault to be used in the CSB contains a 10 by 22 array of storage tubes and has concrete on four sides, the top, and the basemat below. Two MCOs are placed in each storage tube in a vertical column. The operations floor at the top of the tubes is 5-ft-thick concrete. Each tube is a penetration in the operating floor that extends down to the vault floor. An impact absorber approximately 36 in. long is located at the bottom of each tube, and an impact absorber approximately 24 in. long is located between the two MCOs in a tube. The vault walls are closest along the 22-tube side, but over 10 ft distant on the 10-tube side.

The MCOs are removed from the shipping cask and are raised and lowered through the concrete floor using the MHM. When an MCO is removed from or inserted into a storage tube, it is surrounded along its length by the operating deck slab. An MCO reflected by thick concrete is analyzed in Section 6.5. However, the greatest reflection during its movement in the CSB is when the MCO is in the MHM surrounded by stainless steel. A single MCO is modeled with 10 in. of stainless steel directly around and above, with thick concrete below. The MCNP results in Table 6-9 are for normally loaded MCOs inside the MHM. The two cases in Table 6-9 give the k_{eff} for normal, dry MCOs containing air with water vapor at 0.0051 g/cm³ (3 kg of water per MCO).

Cases csb1.7 and csb4.7 (Table 6-9) model dry MCOs containing highest k_{eff} loadings of Mark IA and Mark IV fuel and scrap. For the two types of fuel in a normal arrangement for normal CSB activities, k_{eff} is less than 0.40. This result shows that when the MCO is moved under normal conditions, there is a significant margin for criticality safety. The thick steel shielding in the MHM isolates the MCO neutronically, so inserting or removing an MCO from a storage tube would decrease k_{eff} from the calculated cases by increasing distance and isolation from the other MCOs. As the MCO is lowered into a storage tube, k_{eff} will be affected by the concrete operating deck. The highest k_{eff} occurs when the MCO is lowered 5 ft (i.e., the base of the MCO is even with the bottom of the operating deck). See Cases mhm1.6 through mhm1.10 and mhm4.6 through mhm4.10 in Table 6-9.

Table 6-8. Analysis Results for Normal Multi-Canister Overpack Shipping Casks in the Cold Vacuum Drying Facility.

File ID	Fuel type	MCO water density	Cask annulus density	Comments	Calculation results*		
					k_{calc}	σ_{calc}	k_{eff}
mk1r.2	Mark IA	1.0	1.0	Flooded MCO, annulus	0.8902	0.0010	0.9013
mk4ri.4	Mark IV	1.0	1.0	Flooded MCO, annulus	0.9154	0.0008	0.9265
cvd1.6	Mark IA	0.0051	1.0	Dry MCO, flooded annulus	0.3461	0.0006	0.3571
cvd4.6	Mark IV	0.0051	1.0	Dry MCO, flooded annulus	0.3698	0.0006	0.3808
cvd1.8	Mark IA	0.0051	0.0012	Dry MCO, annulus	0.2799	0.0005	0.2909
cvd4.8	Mark IV	0.0051	0.0012	Dry MCO, annulus	0.3164	0.0005	0.3274

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

$$*k_{eff} = k_{calc} + 0.0004 + ((0.01)^2 + (1.645^2)(\sigma_e^2 + 0.002083^2))^{\frac{1}{2}}$$

as defined in Section 6.1. This value should be less than 0.95.

MCO = multi-canister overpack.

Table 6-9. Multi-Canister Overpack Reactivities in the Canister Storage Building.

Case	Configuration analyzed	k_{calc}	σ_c	k_{eff}^*
csb1.7	Mark IA MCOs in the CSB vault	0.34169	0.00034	0.35268
csb4.7	Mark IV MCOs in the CSB vault	0.36370	0.00038	0.37469
mhm1.6	Mark IA MCO in the MHM	0.28571	0.00046	0.29671
mhm1.8	Mark IA MCO lowered 2.5 ft through the floor	0.28855	0.00049	0.29955
mhm1.10	Mark IA MCO lowered 5 ft through the floor	0.31338	0.00054	0.32439
mhm4.6	Mark IV MCO in the MHM	0.32206	0.00070	0.33309
mhm4.8	Mark IV MCO lowered 2.5 ft through the floor	0.32221	0.00066	0.33324
mhm4.10	Mark IV MCO lowered 5 ft through the floor	0.34229	0.00081	0.35334

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

* $k_{eff} = k_{calc} + 0.0004 + ((0.01)^2 + (1.645)^2(\sigma_c^2 + 0.002083^2))^{1/2}$ as defined in Section 3.2. This value should be less than 0.95.

CSB = Canister Storage Building.

MCO = multi-canister overpack.

MHM = multi-canister overpack handling machine.

6.4.5 Sensitivity Analysis

A number of design variable uncertainties (i.e., assorted fuel assembly lengths, unresolved cask dimensions, and degree of fuel corrosion) affect k_{eff} . These uncertainties are addressed in this section. Base case calculations were performed with conservative assumptions (e.g., the long lengths for Mark IV and Mark IA fuel assemblies, extremes among the unresolved design dimensions, and optimum or full moderator densities). The results of those calculations are listed in Table 6-10 and show the relationship of k_{eff} to several examples of design variable uncertainties.

6.4.5.1 Fuel Length. The Mark IA and Mark IV fuel assemblies were manufactured in discrete lengths. The longest length fuel assembly is 26.1 in. Only 12 of the 26.1-in.-long Mark IA assemblies are stored in the K Basins. The sensitivity studies assumed that the maximum length of the Mark IA assemblies was 20.88 in. The values for k_{eff} for flooded MCOs loaded using these fuel lengths are listed in Table 6-10. Cases ocvd4.1 and ocvd4.2 in Table 6-10 are flooded MCOs loaded with 19.6-in.-long and 14.9-in.-long Mark IA fuel assemblies in the four central baskets. Cases ocvd4.3, ocvd4.4, and ocvd4.5 in Table 6-10 are flooded MCOs loaded with 24.6-in.-long, 23.2-in.-long, and 17.4-in.-long Mark IV fuel assemblies in the three central baskets. The reactivities of these cases show a modest sensitivity to the fuel length variations. The maximum k_{eff} for a Mark IA loaded MCO was 5.5 mk greater than the k_{eff} for the longest-length fuel assemblies (cases ocvd1.1 and ocvd4.1); this difference is within the typical two standard deviations error in the MCNP calculations. The k_{eff} for Mark IV fuel decreased as fuel length decreased when compared to the base case ocvd1.2, with a maximum decrease of about 7 mk for case ocvd4.4.

6.4.5.2 Mass Change Caused by Fuel Corrosion. In Table 6-10, cases ocvd4.11, ocvd4.12, and ocvd4.13; and ocvd4.14, ocvd4.15, and ocvd4.16 involve counteracting situations of fuel mass being lost by corrosion before loading and of fuel mass being added in the lower baskets by corrosion particles of uranium oxide dropping from upper baskets. The second set of cases investigates the effect on k_{eff} of fuel mass redistribution; it was modeled as a conservative uniform mass increase in the central fuel baskets.

Fuel mass will be lost from damaged fuel because of corrosion and because of cleaning of the fuel in the K Basins before the fuel is loaded into the MCOs. The effect on k_{eff} of this fuel mass loss is shown in cases ocvd4.11, ocvd4.12, and ocvd4.13 (Table 6-10), in which a uniform loss of mass in an MCO containing Mark IV fuel was modeled by reducing the fuel density by 10%, 20%, and 30%, respectively, while keeping the fuel assembly dimensions constant. The values for k_{eff} in these cases show a slight and uncertain effect on reactivity for uniform loss at and below 20%. The results show a definite decrease of about 9 mk in k_{eff} corresponding to a decrease of 30% fuel density.

Table 6-10. MCNP Results for Sensitivity Studies. (2 sheets)

File ID	Basket contents							Comments	Calculation results		
	Tier 1	Tier 2	Tier 3	Tier 4	Tier 5	Tier 6			k_{calc}	σ	k_{eff}
ocvd1.1	1.25 wt% scrap	48 Mark IA	48 Mark IA	48 Mark IA	48 Mark IA	1.25 wt% scrap		4 baskets of 49.8 cm long fuel	0.8826	0.0031	0.8947
ocvd4.1	1.25 wt% scrap	48 Mark IA	48 Mark IA	48 Mark IA	48 Mark IA	1.25 wt% scrap		4 baskets of 49.8 cm long fuel	0.8877	0.0036	0.9002
ocvd4.2	1.25 wt% scrap	48 Mark IA	48 Mark IA	48 Mark IA	48 Mark IA	1.25 wt% scrap		4 baskets of 37.8 cm long fuel	0.8808	0.0031	0.8929
ocvd1.2	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	3 baskets of 62.5 cm long fuel			0.8894	0.0023	0.9010
ocvd4.3	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	3 baskets of 62.5 cm long fuel			0.8837	0.0024	0.8954
ocvd4.4	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	3 baskets of 58.9 cm long fuel			0.8817	0.0034	0.8941
ocvd4.5	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	3 baskets of 44.2 cm long fuel			0.8852	0.0028	0.8971
ocvd4.11	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	3 baskets of intact fuel mass reduced by 10%			0.8891	0.0028	0.9010
ocvd4.12	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	3 baskets of intact fuel mass reduced by 20%			0.8883	0.0032	0.9005
ocvd4.13	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	3 baskets of intact fuel mass reduced by 30%			0.8808	0.0029	0.8928
ocvd4.14	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	3 baskets of intact fuel mass increased by 100 kg UO ₂			0.8767	0.0029	0.8887

Table 6-10. MCNP Results for Sensitivity Studies. (2 sheets)

File ID	Basket contents						Comments	Calculation results		
	Tier 1	Tier 2	Tier 3	Tier 4	Tier 5	Tier 6		k_{calc}	σ	k_{eff}
ocvd4.15	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	3 baskets of intact fuel mass increased by 200 kg UO_2		0.8624	0.0030	0.8745
ocvd4.16	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	3 baskets of intact fuel mass increased by 300 kg UO_2		0.8582	0.0027	0.8701
ocvd4.17	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	0.5 in. decrease of cask annulus region		0.8991	0.0028	0.9110
ocvd4.18	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	0.5 in. increase of cask annulus region		0.8829	0.0028	0.8948
ocvd4.19	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	1.0 in. increase of cask annulus region		0.8793	0.0030	0.8914
ocvd4.20	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	1.5 in. increase of cask annulus region		0.8776	0.0036	0.8901
ocvd4.21	0.95 wt% scrap	54 Mark IV	54 Mark IV	54 Mark IV	0.95 wt% scrap	2.0 in. increase of cask annulus region		0.8707	0.0031	0.8828

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSE-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

Fuel mass in each of the three central fuel baskets was assumed to be increased by uranium oxide dropped from corroded Mark IV fuel and Mark IV fuel scrap in the upper baskets. The uranium oxide is redistributed in the interstitial regions between fuel assemblies and in the coolant channels of the fuel assemblies displacing the water. The effects on k_{eff} of this fuel mass increase are shown in cases ocvd4.14, ocvd4.15, and ocvd4.16 (Table 6-10), in which a uniform mass of 100 kg, 200 kg, and 300 kg of UO_2 has been modeled as being distributed in the central three fuel baskets. The reactivities of these cases show a progressive decrease with increasing UO_2 mass when compared with case ocvd1.2 for the Mark IV MCO in Table 6-8, as the UO_2 replaces the water in the fuel basket array.

6.4.5.3 Thickness of Cask Annulus Water. The effect of cask annulus water thickness on k_{eff} also has been investigated. Cases ocvd4.17 through ocvd4.21 in Table 6-10 show k_{eff} for a Mark IV MCO. Case ocvd4.17 assumes no cask annulus gap, and cases ocvd4.18 through ocvd4.21 assume progressive increments of 0.5 in. above the base case thickness of 0.5 in. The results in these cases show that the k_{eff} progressively decreases with increasing cask annulus water thickness, which decreases the reflection of neutrons back into the fuel from the cask wall.

6.4.5.4 Water Level and Density. Explicit calculations treating partially water-filled MCOs have been performed. These cases, shown in Figure 6-10, indicate a slight k_{eff} peak greater than the fully flooded case. The increase in k_{eff} at the peak (above the fully flooded case) is only on the order of the two standard deviations error in the MCNP results but demonstrates the effect of the drained fuel reflecting neutrons into regions where the fuel and scrap are still flooded. The lowest water level shown in Figure 6-10 corresponds to the lower scrap basket being half-flooded; the dry case is so unreactive ($k_{\text{eff}} < 0.40$) that the vertical scale becomes expanded to the point where the shape in the remaining curve becomes difficult to see.

6.4.5.5 Plutonium Distribution. Optimal scrap is treated as unexposed fuel pieces, with no plutonium buildup. One concern has been that plutonium buildup on the exterior surface of fuel elements might preferentially corrode off of the element and contribute to more highly reactive scrap pieces in basin sludge. This question has been addressed for decladding waste streams sent to the Hanford Site tank farms (Rogers et al. 1996). While enhanced plutonium buildup near the outer surface of nuclear fuel (not just Hanford Site reactor fuel) is well-known, the peaking factor (which may be defined as the plutonium concentration at the surface divided by the average plutonium concentration) is not much greater than a factor of two (see Section 6.7.3).

The analyses described in Section 6.7.3 show that the decrease in k_{eff} from ^{235}U burnup is greater than the increase in k_{eff} from ^{239}Pu and ^{241}Pu production because fission products, primarily ^{240}Pu , are also produced and act as poisons.

Enhanced ^{235}U depletion also occurs at the fuel surface. Criticality is affected by both the ^{239}Pu buildup and the ^{235}U depletion. However, fuel corrosion tends to progress axially from damaged ends, and not radially inward. Comparing 1.25 wt% enriched uranium and uranium-plutonium solutions at optimal hydrogen-to-heavy metal atom ratios illustrates the U/Pu ratio required for equivalency.

A parametric study was carried out for solutions of uranium and plutonium as the H/U and U/Pu ratios were varied (Erickson 1994). Table 2 of WHC-SD-NR-CSER-014, *Criticality Safety Evaluation Report for the 100 KE Basin Sandfilter Backwash Pit* (Erickson 1994), contains calculated masses having a minimum k_{eff} of 0.98 as a function of the U/Pu ratio. These results are a minimum because for each U/Pu ratio, the H/U ratio was optimized.

Inspection of Erickson (1994) shows that the U/Pu ratio must be less than 25 for the hemispherical heavy metals safe mass (30.8 kg) to approach the 30 kg of sludge expected to build up in the MCO from the corrosion of the fuel (Reilly 1998). The uranium and plutonium isotopic breakdown assumed in WHC-SD-NR-CSER-014 (Erickson 1994) was based on measurements taken from the K East Basin sandfilter backwash pit and is considered representative of actual uranium and plutonium concentrations currently in the fuel storage basins. Furthermore, the measured U/Pu ratio was in excess of 300 and is expected, from N Reactor production tables (Hedengren and Goldberg 1987), to be 450, which also is consistent with WIMS-E burnup results shown in Section 6.7.3.

Therefore, treating fuel corrosion product sludge as 1.25 wt% UO_2 solution will bound any real corrosion product sludge arising from exposed fuel assemblies. The measured U/Pu data indicate that plutonium buildup is insufficient to yield a value for k_{eff} greater than the assumed 1.25 wt% enriched solution model.

6.4.5.6 Lattice Spacing for Intact Fuel Assemblies. The MCO analyses use a center-to-center hexagonal spacing of 2.8 in. for fuel assemblies. The spacing is established by rings on the bottom of the fuel baskets. Previous analyses have shown that the optimal spacing of N Reactor fuel in water is approximately 3.1 in. (Wittekind 1992). Using a larger spacing would make loading the MCO containers easier, but it also would increase the reactivity of the lattice, because the original design was undermoderated.

The effect of fuel assembly spacing has been analyzed using MCNP for the geometric arrangement in Figure 6-11 using various lattice spacings from 2.65 in. to 2.90 in. The axial arrangement was as shown in CASE1 (Figure 6-7), but with both top and bottom scrap baskets replaced with intact fuel so that the effect would not be overshadowed by the more reactive scrap material. A lattice spacing of 2.90 in. is only possible if the outermost 12 assemblies are removed; otherwise they would impact the inner wall of the MCO. This reduced arrangement is shown in Figure 6-12.

The CASE1 MCNP model assumes an MCO inner diameter of 23.0 in., but the nominal value is only 22.625 in. Assuming this smaller value, the largest possible lattice spacing is 2.8041 in., as noted on Figure 6-13. Although an array of intact assemblies with a lattice spacing of 2.85 in. fits inside the MCNP model (with a 23.0-in. inner diameter), it will not fit inside the nominal 22.625 in. inner diameter.

Removing 12 assemblies to allow lattice spacing of 2.90 in. reduces k_{eff} and does not present a potential problem if the baskets are not fully loaded (see Figure 6-13). The actual lattice spacing of 2.77 in. is conservatively approximated by 2.8 in. With the bottoms of the fuel spaced

at 2.77 in. and the unrestrained tops at 2.8 in., the average spacing would be 2.785 in. The change in k_{eff} for this negligible change (0.015 in.) in lattice spacing is only 0.004, derived from examination of the data shown in Figure 6-13. This effect of increased fuel element spacing at the unrestrained tops of the intact fuel elements is small and will be neglected. Since decreasing the lattice spacing decreases the k_{eff} , ring spacing on the fuel bottom plates of less than 2.8-in. is conservatively covered by this analysis.

6.4.5.7 Partial Fuel Loading. An evaluation was performed to determine the basket loading giving the highest value for k_{eff} . This evaluation used the original design of the Mark IA baskets containing an XXS central pipe and Mark IV fuel baskets with a 3/8 in. thick base plate. Section 6.4.2.4 contains a summary of the results of the same evaluation performed for the redesigned baskets. A full fuel basket is slightly undermoderated and a basket containing one or more empty locations of locations containing only outer elements could approach a more optimal geometry.

The first step was to determine the most reactive loading for an MCO containing both scrap and fuel baskets. New MCNP models were created to reflect the present basket designs using the scrap basket loading limits from HNF-SD-SNF-CSER-010 (Kessler and Peck 2000). Additionally, the Mark IV scrap basket was modeled with nominal dimensions for the basket sides and no central process tube, the most reactive configuration as described in Section 6.4.5.13. Each calculation was performed with only one empty basket location in each fuel basket in the MCO. Table 6-11 shows the results of these analyses.

Figure 6-14 shows the MCNP input used to model the fuel in the basket. Universe 2 is a dummy cell representing the water and universe 3 represents an intact fuel assembly. The numbers on the right and on the bottom may be used to define the location of each assembly. The most reactive configuration is when the assembly in row 3, column 5, is missing. The configurations in Table 6-11 are keyed to this grid.

The last three cases in the table show the results for six assemblies missing from symmetrical locations and for two assemblies missing from alternate locations. These cases show that having more than one assembly missing from a basket lowers the k_{eff} . Table 6-12 shows the results of the same analysis performed for the Mark IV MCO. Unlike the Mark IA MCO, the most reactive state is for the MCO with filled fuel baskets.

The second step was to load only the outer element in the same assembly locations used in previous analyses. Table 6-13 lists the results for the Mark IA MCO and Table 6-14 lists the results for the Mark IV MCO.

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Table 6-13. Mark IA Multi-Canister Overpack with an Outer Element Loaded Only into Selected Locations.

Case	Outer only location	Results		
		k_{calc}	σ	k_{eff}
omk1ao.1	5,5	0.8682	0.0006	0.8793
omk1ao.2	4,5	0.8693	0.0006	0.8804
omk1ao.3	3,5	0.8678	0.0006	0.8788
omk1ao.4	3,6	0.8661	0.0006	0.8772
omk1ao.5	4,6	0.8693	0.0006	0.8803
omk1ao.6	4,4	0.8680	0.0006	0.8790
omk1ao.7	4,5 and 4,7	0.8683	0.0006	0.8793
omk1ao.8	4,5 and 7,8	0.8687	0.0006	0.8798
omk1ao.9	6 outers missing	0.8687	0.0006	0.8797

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

Table 6-14. Mark IV Multi-Canister Overpack with an Outer Element Only Loaded into Selected Locations.

Case	Outer only location	Results		
		k_{calc}	σ	k_{eff}
omk4o.1	5,5	0.9137	0.0006	0.9247
omk4o.2	4,5	0.9129	0.0006	0.9239
omk4o.3	3,5	0.9132	0.0007	0.9243
omk4o.4	3,6	0.9131	0.0006	0.9241
omk4o.5	4,6	0.9151	0.0006	0.9261
omk4o.6	4,4	0.9130	0.0006	0.9240
omk4o.7	5,6	0.9148	0.0006	0.9258

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

For the Mark IA MCO, the most reactive configuration is with an outer element only loaded in a middle row location. This case is also the most reactive when compared to the most reactive case containing an empty location. The value for k_{eff} increases but is still less than the 0.95 limit. The final three cases in Table 6-13 show that k_{eff} decreases if more than one location contains only an outer element. The results for the Mark IV MCO show that the full baskets are the most reactive configuration.

Two additional calculations were performed for the Mark IA MCO to show that if a fuel basket contains a location with only an outer element and an empty location it is still bounded by the maximum case of a single location containing an outer element. Table 6-15 shows these results.

Table 6-15. Mark IA Multi-Canister Overpack with One Location Containing an Outer Element and One Empty Location.

Case	Location	Results		
		k_{calc}	σ	k_{eff}
omklao.8	Outer 4,5 Empty 4,7	0.8678	0.0006	0.8788
omklao.b	Outer 4,5 Empty 7,8	0.8674	0.0006	0.8784

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSE-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

6.4.5.8 Loading Mark IA Outer Elements and Intermediate Mark IA Inner Elements. Baskets are not monitored for changes in neutron count rate as they are loaded. Because the lattice is slightly undermoderated at a pitch of 2.8 in., removal of some fuel could potentially increase k_{eff} . Mark IA outer elements become optimal (have minimal critical mass) at a spacing of 2.9 to 3.0 in. center-to-center in a hexagonal lattice (Schwinkendorf 1995). This is less than the optimal 3.1-in. lattice spacing for intact assemblies. If the fuel baskets are loaded with only Mark IA outer elements, the lattice is closer to optimal spacing and the inner element is replaced by moderator. A series of calculations were performed to determine the effect on k_{eff} of loading Mark IA fuel baskets with only Mark IA outer elements. Starting with the highest k_{eff} reference case described in Section 6.4.2.4, a single fuel basket was modeled as containing only outer elements. The basket was modeled next to each scrap basket and in the center tiers of the MCO. Loading multiple baskets up to, and including, an MCO containing two scrap baskets and four fuel baskets containing only outer elements was also analyzed. The highest k_{eff} , 0.8895, occurs when two fuel baskets containing only outer elements are loaded in tiers 4 and 5 below the upper scrap basket.

Two intermediate cases, shown in Figures 6-15 and 6-16, were modeled with 24 inner elements loaded in the basket of Mark IA outer elements. The first case, CASE1I2, shown in Figure 6-16, was modeled with the inner elements loaded near the inner and outer edges of the fuel region annulus adjacent to the basket perimeter and the insert. The second case, CASE1I1, shown in Figure 6-16, was modeled with the inner elements loaded in the central region of the fuel region annulus. The k_{eff} was 0.8940 for the first case and 0.8893 for the second case. The results of this study show that k_{eff} decreases as a result of inner elements being loaded to the Mark IA baskets.

6.4.5.9 Fuel Temperature. The effect of temperature on k_{eff} was evaluated for both Mark IV and Mark IA fuel assemblies. The WIMS-E lattice code (Gubbins et al. 1982) was used to calculate the lattice k_{∞} and the change in k_{eff} for both Mark IV and Mark IA fuel as a function of temperature, the temperature coefficient. The temperature coefficient may be approximated between two discrete temperatures using the following relationship:

$$\alpha(T) \equiv \left. \frac{\partial \rho}{\partial T} \right|_T \approx \left. \frac{\Delta \rho}{\Delta T} \right|_T = \frac{10^5 (k_{\infty}(T + \Delta T) - k_{\infty}(T - \Delta T))}{k_{\infty}(T + \Delta T)k_{\infty}(T - \Delta T) (2\Delta T)} \text{ pcm/K.}$$

The 10^5 multiplier in the equation above converts the raw k_{eff} , $\Delta \rho$, into percent milli-k (pcm). Figure 6-17 illustrates the lattice k_{∞} and temperature coefficients for both Mark IA and Mark IV fuel. As expected, both N Reactor fuel types have negative temperature coefficients because they contain large percentages of ^{238}U , which has a large negative doppler coefficient. Mark IV fuel exhibits a slightly more negative temperature coefficient than Mark IA fuel because the ^{238}U content is greater. At room temperature (MCNP calculations reported in this chapter were performed at 300 K), the temperature coefficients are approximately equal to -3 pcm/K.

6.4.5.10 Dimensional Tolerance. The effect of dimensional tolerance on k_{eff} was analyzed parametrically by a series of calculations using the WIMS-E code (Gubbins et al. 1982). Table 6-16 contains the radial dimension specifications for Mark IV and Mark IA fuel types (Jack 1988). Table 6-17 shows the variation in lattice k_{∞} as the radial fuel dimensions are varied. Either minimum or maximum dimensions have been selected to arrive at either minimum or maximum uranium fuel region thicknesses. In all cases, the fuel assemblies were placed in an infinite water lattice at optimal spacing. The sensitivity of k_{eff} to radial dimension tolerances is shown to be approximately 1 mk.

6.4.5.11 Enrichment Tolerance. The effect of enrichment tolerance on k_{eff} was analyzed parametrically by a series of calculations using the WIMS-E code (Gubbins et al. 1982). The enrichment tolerance for N Reactor fuel was equal to 0.006 wt% ^{235}U (Gant and Zilar 1977). Table 6-18 shows the variation in lattice k_{∞} as the fuel enrichment is either increased or decreased by 0.006 wt% ^{235}U . In all cases, the fuel assemblies are placed in an infinite water lattice at optimal spacing. The sensitivity of k_{eff} to enrichment tolerance has been shown to be approximately ± 2 mk for a ± 0.006 wt% ^{235}U enrichment variation.

Table 6-16. Radial Dimension Specifications for N Reactor
Mark IV and Mark IA Fuel Assemblies.

	Mark IA (in.)	Mark IV (in.)
Outer element:		
Outer diameter	2.391 - 2.416	2.410 - 2.435
Inner diameter	1.754 - 1.779	1.691 - 1.716
Inner element:		
Outer diameter	1.237 - 1.256	1.267 - 1.286
Inner diameter	0.431 - 0.450	0.473 - 0.492

Table 6-17. Sensitivity of Lattice k-Infinity to Radial Dimension Tolerances.

	Mark IA	Mark IV
Minimum uranium thickness	1.131423	1.062878
Nominal uranium thickness	1.132133	1.062349
Maximum uranium thickness	1.132055	1.061646

Table 6-18. Sensitivity of Lattice k-Infinity to Enrichment Tolerances.

	Mark IA	Mark IV
Minimum uranium enrichment	1.130463	1.060261
Nominal uranium enrichment	1.132133	1.062349
Maximum uranium enrichment	1.133780	1.064408

6.4.5.12 Packing Fraction. The packing fraction is the volume fraction of fuel in the unit lattice. An optimal packing fraction has been used for scrap with both size and spacing varied to maximize k_{eff} . This packing fraction varies according to enrichment but is equal to 0.340 for Mark IV scrap and 0.317 for Mark IA scrap (Kessler and Peck 2000).

6.4.5.13 Scrap Basket Dimensions. The Mark IV scrap basket is limited to a maximum mass of 980 kg (Kessler and Peck 2000). Calculations were performed to show that the k_{eff} is more sensitive to the height of the scrap in the basket than it is to the basket diameter. Two calculations were performed to show this effect. The first modeled the 980 kg of scrap in a basket with a central process tube and no sides, meaning the scrap was contained by the inner wall

of the MCO. This model has a gap of approximately 2.2 cm between the top of the scrap and the base of the next basket. The second modeled the scrap in a basket with the diameter adjusted such that the scrap filled the basket up to the base of the next higher basket. Table 6-19 shows the results of this analysis and the change in k_{eff} .

Table 6-19. Sensitivity Results for Height of Scrap in a Basket.

Case	Configuration	Results		
		k_{calc}	σ	k_{eff}
mk4r.0	Two scrap baskets with three fuel baskets	0.9130	0.0007	0.9241
mk4r.1	No basket sides	0.8893	0.0007	0.9003

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

No sensitivity studies were performed to determine the effects of either radial or axial movement of the scrap baskets. The sides of the scrap baskets are designed to maintain contact with the inner surface of the MCO. In addition, the center post of the Mark IA basket and central pipe of the Mark IV basket are both designed to fit into the base of the next basket. Both these features limit the radial movement of the baskets to approximately 0.02 in. Because the Mark IV scrap basket is assumed to be full, axial movement may only occur if both its base plate and the base plate of the fuel basket above it were 1.23 in. thick, the minimum thickness allowed. The total movement would be 0.04 in. Movements as small as these have only a marginal effect on k_{eff} . Addition of a water gap, as shown in Table 6-19, will lower k_{eff} because the water is both a reflector and a moderator. This water gap therefore reduces the neutron interaction of the scrap basket with the basket above it.

6.5 CONTINGENCY ANALYSIS

6.5.1 Summary and Conclusions

Summaries of the normal and off-normal conditions of the MCO are provided in Tables 6-20 and 6-21. The off-normal conditions listed in Table 6-20 are those resulting from a single contingency and must meet the single-contingency criterion while those listed in Table 6-21 require more than one unlikely and independent contingency to occur and are not required to have a k_{eff} of less than 0.95.

Table 6-20. Normal and Off-Normal Multi-Canister Overpack Conditions
Involving a Single Contingency.

Facility	Normal condition	Off-normal condition	Section
Cold Vacuum Drying Facility	MCO flooded inside transfer cask	Mark IA fuel and scrap misloaded into Mark IV baskets	6.5.3.1
		Scrap baskets misloaded	6.5.3.2
Canister Storage Building	MCO dry inside transfer cask	Vertical drop	6.5.2.1
		Horizontal drop	6.5.2.2
	MCO dry inside MCO handling machine	MCO flooded	6.5.4.1
	MCO dry being transferred into storage tube	MCO flooded	6.5.4.2
	MCO dry double stacked inside storage tubes	Vault flooded	6.5.4.3
		Storage tube flooded	6.5.4.4
		MCO flooded	6.5.4.5
		Misloaded fuel and scrap	6.5.4.6
K Basin	MCO flooded inside the transfer cask	Vertical drop and MCO flooded	6.5.2.1
		Horizontal drop and MCO flooded	6.5.2.2
		Mark IA insert offset and MCO flooded	6.5.2.3

MCO = multi-canister overpack.

Table 6-21. Normal and Off-Normal Multi-Canister Overpack Conditions
Involving Multiple Contingencies.

Facility	Normal condition	Off-normal condition	Section
Canister Storage Building	MCO dry double stacked inside storage tubes	MCO flooded and fuel and scrap misloaded	6.5.4.5
			6.5.4.6

MCO = multi-canister overpack.

Three categories of contingencies were addressed as part of the criticality safety analysis: drops, misloadings, and flooding in the CSB. The most significant off-normal condition is the MCO design basis drop accident. The flooded drop accident in the CVDF is not considered credible because the MCO is not lifted and the transfer cask cannot spill or fall from the transporter during design basis natural phenomena (seismic, tornado) (Chenault 1997b). Drop accidents and the credible forces associated with them are discussed in Section 2.2.6.3 and Chapter 3.0. The MCO shield plug assembly and Mark IA fuel and scrap basket baseplates and center posts are safety class for the design basis accident so that the MCO contents will not spill out, displace the Mark IA center post beyond its acceptable range, or allow the fuel and scrap baskets to collapse onto each other. Dropping a loaded cask is assumed to rubblize the intact fuel in the MCO fuel baskets, drop the baskets into a stack separated by 1.25-in. plates, and to offset the central pipe insert 2 in. off center, allowing rubble and scrap closer to the centerline of the MCO. Rubblized fuel is modeled as optimized scrap completely filling the MCO. Scrap baskets and fuel rubble are assumed to expand to the inside diameter of the MCO, 23.25 in. The calculations identified in Table 6-22 model rubble created by a drop as optimally sized and spaced scrap rods, including cladding.

Table 6-22. Reactivity for Multi-Canister Overpack Drops.

Case	Most reactive MCO configuration	k_{calc}	σ_c	k_{eff}
mk1ad.4	Six Mark IA fuel baskets, packing fraction is 0.317	0.9057	0.0010	0.9168
mk4d.5	Two Mark IV scrap baskets, three fuel baskets, packing fraction is 0.340	0.9289	0.0006	0.9399
mk1ad.8	Six Mark IA fuel baskets, central pipe offset by 2 in., packing fraction is 0.317	0.9134	0.0009	0.9244
spr.2	SPR fuel	0.8752	0.0030	0.8872

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

SPR = single-pass reactor.

Mark IA fuel (1.25 wt% ^{235}U enrichment) baskets have a center post, discussed in Section 6.2.2, to exclude fuel from this space during flooded conditions. The drop accident could offset that post. The post has been designed to allow less than a 2-in. offset in the drop accident during flooded conditions (Goldmann 2000b). At that offset, k_{eff} is less than 0.95.

An MCO containing Mark IV fuel was analyzed in Section 6.5.3.1 for misloading Mark IA fuel in a Mark IV fuel basket containing Mark IV fuel. For both intact fuel elements and scrap,

the results were within allowable limits for MCOs alone, in a cask, or stored in the CSB for the credible misloading of one canister of Mark IA fuel or scrap. The other misloading cases require multiple misloadings and are provided for information only.

For dry MCOs, the k_{eff} is low. Adding water will increase the k_{eff} . Analysis of both adding water and a single contingency of misloading an MCO with higher enrichment fuel does not exceed allowable limits. Transfer of an MCO between the MHM and the storage tube was analyzed in Section 6.4.4. Considering the structural materials as reflectors, the k_{eff} of the MCO was within limits for both dry and flooded conditions.

The analysis in Section 6.5.3.3 shows that for a third scrap basket in the flooded MCO in its cask at the CVDF, the k_{eff} is within allowable limits. Receiving an MCO containing a third scrap basket is considered to be the worst contingency expected for the CVDF.

6.5.2 Multi-Canister Overpack Drop

This section provides a quantitative evaluation of factors that were considered for the final design models of the dropped MCO. Drop accidents involving flooded MCOs may only occur during removal from the K Basins and shipment to CVDF. No credible drop accidents have been identified in Section 3.3.2.2 of the CVDF SAR (Pili-Vincens 1998) that would involve a flooded MCO. The flooded drop accident at the CVDF is, therefore, a beyond design basis event. The credible drop accidents identified in Section 3.3.2.2 of the CSB SAR (Garvin 1997a) are a shipping cask drop, an MCO drop in the MHM, and an MCO drop into the storage tube. These drop accidents involve an MCO that has been dried at the CVDF. The design basis accident loadings are discussed in Section 2.2.6.3.

The MCO design includes safety-class features to maintain fuel and scrap in the MCO following design basis drop accidents. A HEPA filter assembly is included as part of the safety-class MCO plug and is protected by a 1.5 in. thick safety-class filter guard plate, which is located approximately 0.3 in. above the top basket after installation. The plug is designed to prevent SNF from being released from the MCO in a design basis accident. As an integral part of the shield plug, this plate limits the volume available for the rubblized fuel and scrap to fill following a drop accident. The safety-class base plates and center posts of the Mark IA baskets also provide geometry control. Although the central pipe and baseplates are not safety class for the Mark IV baskets, the steel mass of the basket base plate will remain in the MCO following the drop and is included in the model. The basket sides are not included in any drop models (see Figure 6-18). Table 6-22 lists the maximum k_{eff} for both the Mark IA and Mark IV MCOs following a drop with both the fuel and scrap at optimum packing fractions. The optimum packing fraction for Mark IA scrap is 0.317, and the optimum packing fraction for Mark IV scrap is 0.340 (Kessler and Peck 2000). Because of the scrap loading limit for the Mark IA scrap baskets, an MCO containing six fuel baskets with the fuel rubblized to scrap is the most reactive configuration. The most reactive Mark IV configuration is the two scrap baskets, three fuel baskets case. Since the MCOs are full, final orientation (i.e., vertical, or horizontal from a "slap down") does not affect reactivity. Case mk1ad.8 shows the effect of displacing the central pipe by 2 in. for the most

reactive Mark IA drop configuration. Refer to HNF-SD-SNF-CSER-005 (Kessler 2000) for a complete discussion.

The results for dry MCO drops at CSB are listed in Table 6-23 as cases i2.IA.00 and i2.IV.00 for the Mark IA and Mark IV MCOs, respectively. The value of k_{eff} for both cases is below 0.4. These analyses were not conducted using the updated MCO basket designs; they are bounded by the drop of flooded MCOs and are included for information only.

The geometry of an MCO with SPR fuel is shown as spr.2 in Figure 6-8. The aluminum-clad SPR fuel has not been observed to have its mechanical integrity degraded by cladding failure and corrosion as is the case with N Reactor fuel. Therefore, the fuel was modeled as representative aluminum-clad uranium cylinders with lattice spacing corresponding to optimum moderation. The inside diameters of the scrap baskets containing the SPR fuel were expanded to the maximum limit of 23.25 in. The center insert was offset to its limit of 2 in. (Goldmann 2000b).

The enrichments of the uranium cylinders representing depleted, natural, and 0.95 wt% ^{235}U enriched fuels were modeled as the limiting 0.95 wt% ^{235}U cylinders. The mass of unidentified SPR fuel was modeled as a cylinder of 2.1 wt% ^{235}U enriched uranium in the center of the tier four basket. This location has the highest neutron importance. The results of this calculation, shown as spr.2 in Table 6-22, are below 0.95.

6.5.3 Cold Vacuum Drying Facility

6.5.3.1 Mark IA Fuel and Scrap Loaded into Mark IV Baskets. This section and Table 6-24 present the effects of misloading MCOs with fuel and scrap. Each MCO analyzed contained five Mark IV baskets fully flooded and fully reflected by the cask steel, and the annulus was flooded. For the CSB, this represents the contingencies of flooding and multiple misloadings; for the CVDF, this represents multiple misloading contingencies.

One or more baskets is misloaded for each case in Table 6-24. The credible misloading scenario is one canister containing 14 Mark IA fuel assemblies loaded into a Mark IV fuel basket or the mass equivalent to 14 Mark IA fuel assemblies (233 kg) as 1.15 wt% ^{235}U scrap in a Mark IV scrap basket (Kessler 2000). An inventory exists for each canister in the K Basins and intact fuel assemblies can be visually identified as either Mark IA or Mark IV. Therefore, the only credible misloading is one canister of Mark IA fuel or scrap, equivalent to 233 kg. For this analysis the baskets containing Mark IV fuel were modeled with the largest Mark IV assemblies.

Table 6-23. Analysis Results for Multi-Canister Overpack Drop Accidents at Canister Storage Building.

File ID	Fuel type ^a	Drop position	Basket spacing ^b (in.)	MCO interior	Calculation results ^c		
					k_{calc}	σ_{calc}	k_{eff}
i2.IA.00	Mark IA	Vertical	0.375	Dry	0.3193	0.0015	0.3306
i2.IV.00	Mark IV	Vertical	0.375	Dry	0.3272	0.0015	0.3385
i3.IA.00	Mark IA	Horizontal	0.375	Dry	0.3125	0.0018	0.3239
i3.IV.00	Mark IV	Horizontal	0.375	Dry	0.3266	0.0007	0.3376

Notes: All MCOs are loaded normally with one scrap basket in the top and bottom location.

The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

^aMark IV type neglects the center post, which excludes center assembly in array. Mark IA has a 6.6-in.-diameter steel insert, excluding assemblies from seven central positions.

^bSpacing between top of scrap or rubble and the bottom of scrap or rubble in the basket above.

^c $k_{eff} = k_{calc} + 0.0004 + ((0.01)^2 + (1.645^2)(\sigma_c^2 + 0.002083^2))^*$ as defined in Section 6.1. This value should be less than 0.95.

MCO = multi-canister overpack.

Table 6-24. Basket Misload Analysis for Mark IV Multi-Canister Overpacks.

Case	MCO configuration	k_{calc}	σ_c	k_{eff}^*
mk4rl.1	14 Mark IA fuel assemblies in the center of the tier 2 fuel basket	0.9131	0.0009	0.9242
mk4rl.2	14 Mark IA fuel assemblies in the center of the tier 3 fuel basket	0.9129	0.0009	0.9240
mk4rl.3	14 Mark IA fuel assemblies in the center of the tier 4 fuel basket	0.9112	0.0008	0.9222
mk4rl.4	14 Mark IA fuel assemblies in the outer row of the tier 2 fuel basket	0.9136	0.0009	0.9247
mk4rl.5	14 Mark IA fuel assemblies in the outer row of the tier 3 fuel basket	0.9123	0.0008	0.9233
mk4rl.6	14 Mark IA fuel assemblies in the outer row of the tier 4 fuel basket	0.9127	0.0010	0.9238
mk4rc.13	Mass of 14 Mark IA outer elements in the center of the lower scrap basket	0.9133	0.0009	0.9244
mk4rc.14	Mass of 14 Mark IA outer elements in the center of the upper scrap basket	0.9149	0.0010	0.9259

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

* $k_{eff} = k_{calc} + 0.0004 + ((0.01)^2 + (1.645)^2(\sigma_c^2 + 0.002083^2))^{1/2}$ as defined in Section 3.2. This value should be less than 0.95.

MCO = multi-canister overpack.

Most of the fuel enriched to greater than 0.95 wt% ^{235}U is stored in the K West Basin. Only a small amount is stored in the K East Basin. The K West Basin does not have a significant amount of sludge on the basin floor, nor does the sludge contain a significant amount of fissile material. So the misloaded fuel or scrap must come from a misloaded or mislabeled canister. Because the canister was originally filled with whole fuel assemblies, the maximum enrichment would be 1.15 wt% ^{235}U . For criticality analyses, scrap from Mark IA fuel is modeled as 1.25 wt% ^{235}U rods. A single contingency was analyzed in which one fuel canister containing Mark IA scrap, with the mass equivalent of 14 fuel assemblies, 233 kg, was misidentified and handled as though it were 0.95 wt% ^{235}U scrap. The mass was inserted into one of the Mark IV scrap baskets in an otherwise normally loaded Mark IV MCO (Kessler 2000). The k_{eff} was 0.926, which is well below the allowable limit of 0.95 (case mk4rc.14, Table 6-24).

6.5.3.2 Flooded Multi-Canister Overpack with a Drained Shipping Cask Annulus. This contingency assumed that the cask annulus is drained and the MCO loading is correct. Calculations cvd1.7 and cvd4.7 in Table 6-25 show that a flooded MCO in a shipping cask with a drained annulus has a higher k_{eff} than a flooded MCO in a shipping cask with a flooded annulus. Removal of the annulus water allows more neutrons to be reflected from the cask back into the MCO thus increasing k_{eff} . The results listed in Table 6-25 show that the k_{eff} is less than 0.95 for both Mark IA and Mark IV MCOs.

Table 6-25. Calculation Results for Multi-Canister Overpacks Containing Long-Length Mark IA Fuel Assemblies in Mark IV Baskets with a Drained Annulus.

Case	MCO configuration	k_{calc}	σ_c	k_{eff}^a
cvd1.7	2 scrap baskets and 4 partially loaded fuel baskets in a Mark IA MCO	0.90154	0.00096	0.91263
cvd4.7	2 scrap baskets and 3 partially loaded fuel baskets in a Mark IV MCO	0.92152	0.00088	0.93259
18rc.15 ^b	Mark IA assemblies in the center of the lower basket	0.92340	0.00084	0.93446
mk1rs.1	Two scrap baskets in the bottom two tiers with a fuel basket in the top	0.89278	0.00097	0.90387
mk1rs.2	Two scrap baskets in the bottom two tiers with a scrap basket in the top	0.89063	0.00098	0.90172
mk4rs.1	Two scrap baskets in the bottom two tiers with a fuel basket in the top	0.92256	0.00082	0.93362
mk4rs.2	Two scrap baskets in the bottom two tiers with a scrap basket in the top	0.92040	0.00096	0.93149

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

^a $k_{\text{eff}} = k_{\text{calc}} + 0.0004 + ((0.01)^2 + (1.645)^2(\sigma_c^2 + 0.002083^2))^{1/2}$ as defined in Section 3.2. This value should be less than 0.95.

^bFuel baskets contain an inner element location in position 4,5.

^cFuel baskets contain an inner element location in position 3,6.

MCO = multi-canister overpack.

6.5.3.3 Scrap Baskets Loaded in the Wrong Tier and Extra Scrap Baskets. Contingency case reactivities for misloaded scrap in the flooded MCO and transfer cask are shown in Table 6-25. Cases mk4rs.2 and mk1rs.2 are situations in which the limit of two scrap baskets per MCO is assumed to be inadvertently exceeded by the loading of a third scrap basket. For both Mark IA and Mark IV fuel, the k_{eff} is greatest when the misloaded scrap basket is adjacent to an end scrap basket. These are less than 0.95.

6.5.4 Canister Storage Building

Table 6-8 has shown that the storage of dry MCOs in the CSB results in significantly subcritical conditions. Flooding the MCO, the storage tube, or the vault could raise the k_{eff} to a level of concern. The CSB design has excluded a sprinkler system and has no other piped-in water. During interim storage, the MCOs are sealed. Flooding the MCOs while in storage is not considered credible. However, a flooded MCO may be delivered to the CSB, although this event would require multiple violations of shipping and receiving procedures at the CVDF and CSB. These effects on k_{eff} are calculated to show compliance with the 0.95 limit. The k_{eff} also is calculated for a misloaded MCO, and for moving an MCO in the MHM and into and out of the storage tubes through the concrete floor.

6.5.4.1 Handling a Flooded Multi-Canister Overpack with the Multi-Canister Overpack Handling Machine. Handling a flooded MCO with the MHM and loading it into a storage tube has been analyzed and shown to be acceptable (Kessler 2000). The maximum k_{eff} is for the MCO in the MHM prior to its being lowered into the tube. Table 6-26 shows the results for both Mark IA and Mark IV MCOs and Figure 6-19 shows the MCNP model.

Table 6-26. Flooded Multi-Canister Overpacks in the Multi-Canister Overpack Handling Machine and Lowered through the Canister Storage Building Floor.

Case	MCO initial configuration	k_{calc}	σ_c	k_{eff}^*
mhm1.7	Flooded Mark IA MCO in the MHM	0.8992	0.0010	0.9103
mhm4.7	Flooded Mark IV MCO in the MHM	0.9784	0.0012	0.9300

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

* $k_{\text{eff}} = k_{\text{calc}} + 0.0004 + ((0.01)^2 + (1.645)^2(\sigma_c^2 + 0.002083^2))^{1/2}$ as defined in Section 3.2. This value should be less than 0.95.

MCO = multi-canister overpack.

MHM = multi-canister overpack handling machine.

6.5.4.2 Vault Flooding. The CSB is designed with no water lines, and the only access to the vault space between the storage tubes is the two stacks that allow natural convection circulation for cooling the space. However, to show the conservative nature of the CSB, the contingency of water flooding the vault is analyzed using the infinite array model. The results are shown in Table 6-27. Progressively filling the vault to a quarter full (the bottom MCO is half submerged), and to half full (the bottom MCO is completely submerged), lowers the k_{eff} to 0.3861 and 0.3807, as shown in cases oc2.1 and oc2.2, respectively. Case oc2.8 has an even lower value of 0.3447 for the vault fully flooded (both MCOs submerged). The greater the flooding of the vault, the greater the neutronic isolation of each storage tube. The progressive flooding of a vault decreases the overall k_{eff} of the array.

Another way to increase the k_{eff} of an array of undermoderated elements is to insert water as a vapor or mist between the array elements. The infinite horizontal array model of CSB storage tubes containing two normally loaded MCOs was run with water densities from 0.0005 to 1.0 g/cm³ between the tubes. Figure 6-20 plots this data for internally dry MCOs, and Figure 6-21 plots this data for internally flooded MCOs. The peak in k_{eff} , 0.3968, occurs at 0.008 g/cm³ for internally dry MCOs. Case ocsb05 (Table 6-27) shows the results of the 10 by 22 by 2 hexagonal array model of normally loaded Mark IV fuel in the MCOs with 0.008 g/cm³ of intertube moisture. Thus, the actual k_{eff} for the dry MCO in the CSB storage tube is highly subcritical.

Figure 6-20 also shows that for water densities greater than 0.008 g/cm³ in the vault, the k_{eff} drops to that for a single fully water-reflected MCO. This section also showed that for flooding of the vault space, the k_{eff} was below 0.4 (see Table 6-27). For this degree of subcriticality, even at optimum intertube water density, no restriction on fire fighting is necessary to restrict water from the vault for criticality control.

6.5.4.3 Flooding the Canister Storage Building Storage Tube. Table 6-28 shows that flooding the 1.5-in. gap between the MCO and storage tube has no significant effect on the k_{eff} of dry MCOs in the CSB storage tubes. Comparing case oc1.3b, for vapor in the gap, with oc2.11b, for full density water in the gap, shows all k_{eff} values are between 0.41 and 0.43. Each case uses 0.008 g/cm³ of water in the intertube space. Case oc2.14b models the gap filled with half density water and also shows a k_{eff} value in the 0.41 to 0.43 range. These cases modeled Mark IV fuel. Mark IA fuel is used in case oc2.10ba with full density water in the gap and the k_{eff} value is about 0.41. These results indicate that for dry MCOs, the possibility of flooding storage tubes is not a criticality concern.

6.5.4.4 Interspersed Moderation. As water moderation is decreased inside the MCO, the k_{eff} of the system decreases rapidly, as shown in Figure 6-22. The k_{eff} is insensitive to changes in water density in between MCO tubes in the CSB. As the water between MCO containers is reduced (with water density held constant at 1.0 g/cm³ inside the MCO), the k_{eff} does not change significantly, compared with the 95% confidence intervals of the results themselves. These results are shown in Figure 6-23.

Table 6-27. Analysis Results for Canister Storage Building Vault Flooding.

File ID	Fuel type	Payload cluster		Water density (g/cm ³)		Calculation results ^a		
		Number of baskets	Assemblies per basket	MCO interior	Between storage tubes and outside MCOs	k_{calc}	σ_{calc}	k_{eff}
oc2.1	Mark IV ^b	1 top	0.95 wt% ²³⁵ U scrap	0.0051	1.00 (Vault flooded to one-quarter height)	0.3764	0.0016	0.3877
		3 middle 1 bottom	54 Mark IV 0.95 wt% ²³⁵ U scrap					
oc2.2	Mark IV ^b	1 top	0.95 wt% ²³⁵ U scrap	0.0051	1.00 (Vault flooded to half height)	0.3711	0.0014	0.3823
		3 middle 1 bottom	54 Mark IV 0.95 wt% ²³⁵ U scrap					
oc2.8	Mark IV ^b	1 top	0.95 wt% ²³⁵ U scrap	0.0051	1.00 (Vault flooded to full height)	0.3348	0.0020	0.3463
		3 middle 1 bottom	54 Mark IV 0.95 wt% ²³⁵ U scrap					
ocsb05	Mark IV ^c	1 top	0.95 wt% ²³⁵ U scrap	0.0051	0.008 (Vapor)	0.3874	0.0008	0.3985
		3 middle 1 bottom	54 Mark IV 0.95 wt% ²³⁵ U scrap					

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

^a $k_{eff} = k_{calc} + 0.0004 + ((0.01)^2 + (1.645^2)(\sigma_c^2 + 0.002083^2))^{\frac{1}{2}}$ as defined in Section 6.1. This value should be less than 0.95.

^bInfinite array.

^cModeled using 10 x 22 x 2 array of MCOs in CSB storage tubes with concrete walls, floors and walls. Fuel density corresponds to the actual 10 x 22 x 2 hexagonal array of CSB.

MCO = multi-canister overpack.

Table 6-28. Analysis Results for Canister Storage Building Storage Tube Flooding.

File ID	Fuel type	Payload cluster		Water density (g/cm ³)		Calculation results ^a		
		Number of baskets	Assemblies per basket	MCO interior	Between storage tubes and outside MCOs	k_{calc}	σ_{calc}	k_{eff}
oc1.3b	Mark IV ^b	1 top	0.95 wt% ²³⁵ U scrap	0.0051	0.008 (vapor)	0.4114	0.0020	0.4229
		3 middle 1 bottom	54 Mark IV 0.95 wt% ²³⁵ U scrap					
oc2.11b	Mark IV ^b	1 top	0.95 wt% ²³⁵ U scrap	0.0051	0.008 (vapor) between storage tubes 1.0 outside MCOs	0.4042	0.0026	0.4160
		3 middle 1 bottom	54 Mark IV 0.95 wt% ²³⁵ U scrap					
oc2.14b	Mark IV ^b	1 top	0.95 wt% ²³⁵ U scrap	0.0051	0.008 (vapor) between storage tubes 0.5 outside MCOs	0.4109	0.0020	0.4224
		3 middle 1 bottom	54 Mark IV 0.95 wt% ²³⁵ U scrap					
oc2.10ba	Mark IA ^b	1 top	1.25 wt% ²³⁵ U scrap	0.0051	0.008 (vapor) between storage tubes 0.5 outside MCOs	0.4008	0.0024	0.4125
		4 middle 1 bottom	48 Mark IV 1.25 wt% ²³⁵ U scrap					

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

^a $k_{eff} = k_{calc} + 0.0004 + ((0.01)^2 + (1.645^2)(\sigma_c^2 + 0.002083^2))^{\frac{1}{2}}$ as defined in Section 6.1. This value should be less than 0.95.

^bInfinite array.

MCO = multi-canister overpack.

It should be noted that flooding a storage tube is not considered a credible contingency. The floor plug is sealed, and water and water lines are excluded from the vault operating floor and area. An unusual occurrence would have to happen to bring water onto the operating floor.

6.5.4.5 Flooded Multi-Canister Overpacks. An MCO could be delivered to the CSB that had not been dried. A single flooded MCO is considered a credible contingency. Analysis results in Table 6-29 show that the entire array of MCOs in the CSB could be flooded and the facility would still be within acceptable limits. The k_{eff} of a single flooded MCO in a storage tube will be less than that of the full array of flooded MCOs. The model uses an infinite array of normally loaded MCOs with the internal nonmetal spaces flooded. The first two cases, oc2.12 and oc2.12b, have 0.008 g/cm³ water in the storage tube and intertube water densities of 0.0012 g/cm³ and 0.008 g/cm³, respectively. The k_{eff} values are 0.9221 and 0.9120. When full density water is put in the storage tubes, cases oc2.13 and oc2.13b with 0.0012 g/cm³ and 0.008 g/cm³ of water in the intertube space, the results are lower, 0.8887 and 0.8952. These trends for partial water densities in the vault space are shown in Figure 6-21, which shows the k_{eff} for flooded MCOs loaded with Mark IV fuel in the CSB array of storage tubes for a range of water densities between tubes. The curve rises only marginally, to a maximum of 0.93 at a water density of 0.002 g/cm³. This figure shows that an infinite array of fully flooded MCOs loaded with Mark IV fuel in the CSB array will be within allowable limits for all values of interspersed moderation between the storage tubes.

These results show that for flooded storage tubes containing normally loaded MCOs, the k_{eff} is less than 0.95 and fire fighting does not have to be restricted.

The entries in Table 6-30 all have flooded MCOs and storage tubes. All MCOs are flooded. In three of these cases, the vault space also is flooded. In all cases, the values for k_{eff} are below 0.9. For these conditions, the contingency of flooding raises the k_{eff} , but not above allowable limits. The last two cases, oc3.5c and oc3.6c, present the results of more than two contingencies and are included for understanding the consequences of gross misloading.

6.5.4.6 Multi-Canister Overpacks with Misloaded Fuel. Several MCNP calculations have shown that the CSB is significantly subcritical even for loading Mark IA fuel and scrap in all the Mark IV baskets in all MCOs stored in the CSB as long as the MCOs are not flooded. The model has a 23.0-in. fuel basket diameter, 1.25 wt% ²³⁵U scrap in the top and bottom Mark IV baskets, 54 Mark IA intact fuel assemblies in each of the three center Mark IV baskets, and two MCOs vertically in an infinite array of CSB tubes. With all tubes and intertube spaces dry, the k_{eff} equals 0.3787. If full density water is added between the CSB tubes and inside the CSB tubes outside the MCOs (interior of MCOs dry), the k_{eff} decreases. Even for flooding the CSB vaults, a misloaded MCO is significantly less reactive than the allowed limits as long as the MCO interior is dry. This is true even for the multiple contingencies (i.e., completely misloading each MCO in the CSB). The design of the CSB exceeds the requirements of the double contingency principle.

Table 6-29. Analysis Results for Multi-Canister Overpack Internal Flooding.

File ID	Fuel type	Payload cluster		Water density (g/cm ³)		Calculation results ^a		
		Number of baskets	Assemblies per basket	MCO interior	Between storage tubes and outside MCOs	k _{calc}	σ _{calc}	k _{eff}
oc2.12	Mark IV ^b	1 top	0.95 wt% ²³⁵ U scrap	1.0	0.0012	0.9119	0.0025	0.9236
		3 middle	54 Mark IV					
		1 bottom	0.95 wt% ²³⁵ U scrap					
oc2.12b	Mark IV ^b	1 top	0.95 wt% ²³⁵ U scrap	1.0	0.008	0.9016	0.0027	0.9135
		3 middle	54 Mark IV					
		1 bottom	0.95 wt% ²³⁵ U scrap					
oc2.13	Mark IV ^b	1 top	0.95 wt% ²³⁵ U scrap	1.0	0.0012 between storage tubes 1.0 outside MCOs	0.8782	0.0029	0.8902
		3 middle	54 Mark IV					
		1 bottom	0.95 wt% ²³⁵ U scrap					
oc2.13b	Mark IV ^b	1 top	0.95 wt% ²³⁵ U scrap	1.0	0.008 between storage tubes 1.0 outside MCOs	0.8849	0.0026	0.8967
		3 middle	54 Mark IV					
		1 bottom	0.95 wt% ²³⁵ U scrap					

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

^ak_{eff} = k_{calc} + 0.0004 + ((0.01)² + (1.645²(σ_c² + 0.002083²))^{1/2} as defined in Section 6.1. This value should be less than 0.95.
^bInfinite array.

MCO = multi-canister overpack.

Table 6-30. Analysis Results for Normally Loaded Multi-Canister Overpacks Flooded in the Canister Storage Building.

File ID	Fuel type ^a	Payload cluster		Water density (g/cm ³)		Calculation results ^b		
		Number of baskets	Assemblies per basket	MCO interior	Between storage tubes	k_{calc}	σ_{calc}	k_{eff}
csb1	Mark IV	1 top	0.95 wt% ²³⁵ U scrap	1.0	1.0 (full density)	0.8716	0.0023	0.8832
		3 middle 1 bottom	54 Mark IV 0.95 wt% ²³⁵ U scrap					
csb3	Mark IA	1 top	1.25 wt% ²³⁵ U scrap	1.0	1.0 (full density)	0.8196	0.0020	0.8311
		4 middle 1 bottom	48 Mark IA 1.25 wt% ²³⁵ U scrap					
oc3.5c	Mark IV	1 top	0.95 wt% ²³⁵ U scrap	0.0051	.008 (mist)	0.4110	0.0019	0.4224
		3 middle 1 bottom	54 Mark IV 54 Mark IA 54 Mark IV 0.95 wt% ²³⁵ U scrap					
oc3.6c	Mark IV	1 top	0.95 wt% ²³⁵ U scrap	1.0	.008 (mist)	0.9030	0.0019	0.9144
		3 middle 1 bottom	54 Mark IV 54 Mark IA 54 Mark IV 0.95 wt% ²³⁵ U scrap					

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

^aMark IV type has a center post, which excludes one center assembly in array, but is much smaller than the Mark IA type with a 6.6-in.-diameter steel insert that excludes seven central assemblies. MCOs in a 10 x 22 lattice with two MCOs at each storage tube. One-foot-thick concrete walls are modeled on four sides and the floor.

^b $k_{eff} = k_{calc} + 0.0004 + ((0.01)^2 + (1.645^2)(\sigma_c^2 + 0.002083^2))^{\frac{1}{2}}$ as defined in Section 6.1. This value should be less than 0.95.

MCO = multi-canister overpack.

Calculations have been done for a case of loading one K Basin canister of Mark IA fuel into a Mark IV scrap basket, and for a case of loading a Mark IV scrap basket in place of an intact fuel basket. A third scrap basket is modeled next to the top scrap basket in the bottom MCO in a storage tube. This puts three scrap baskets as close to one another as possible with a single contingency. The results of the analysis are shown in Table 6-31. The k_{eff} is 0.4232 for case oc3.1b, which uses an intertube optimal water density of 0.008 g/cm^3 . The value is lower, 0.4035, for case oc3.1, which uses an intertube water density of 0.0012 g/cm^3 . For a second contingency, the MCO being flooded, the k_{eff} is 0.9349, as shown for case oc3.2. This result shows that for multiple contingencies, the k_{eff} is still less than 0.95. The analysis is conservative in that the bottom MCO in each tube in the array has an extra scrap basket and the array is modeled as an infinite array.

Another contingency is for a single canister of Mark IA fuel to be misplaced and mislabeled as Mark IV fuel in K Basins. This canister is loaded into a Mark IV scrap basket, so a scrap basket intended for 0.95 wt% ^{235}U enriched scrap is loaded with the mass equivalent of 14 assemblies, 233 kg, of 1.25 wt% ^{235}U enriched scrap, and the rest of the basket is loaded with 0.95 wt% ^{235}U Mark IV scrap. Since scrap baskets are more reactive than intact fuel baskets, this case bounds other cases of a single canister of Mark IA fuel in Mark IV baskets. The k_{eff} value is 0.4228 for case oc3.3b, which uses an intertube optimal water density of 0.008 g/cm^3 (Table 6-21). The value is lower, 0.4071, for case oc3.3, which uses an intertube water density of 0.0012 g/cm^3 . Adding a second contingency, the MCO being flooded, the k_{eff} is 0.9229, as shown in case oc3.4b, which has an intertube water density of 0.008 g/cm^3 . This result shows that for multiple contingencies, the k_{eff} is still less than 0.95.

6.5.4.7 Loading a Multi-Canister Overpack in a Tube with no Intermediate Impact Absorber. It is possible to load the second MCO in a storage tube without installing the intermediate impact absorber. For this situation, the CSB array was modeled containing one tube with two MCOs and no intermediate impact absorber and with the entire array of MCOs with no impact absorbers. The model used for these calculations was updated to include the lower and intermediate impact absorber, the latest basket and MCO dimensions, correct mass limits for the scrap baskets, and the full length of the MCO as discussed in Section 6.3. All the arrays of drained MCOs were analyzed and the results reported in Table 6-32. The results show that the presence of the intermediate impact absorber has no effect on k_{eff} for drained MCOs. This is because of the space and plug above the top scrap basket, which isolates the top and bottom MCOs from each other.

6.5.4.8 Flooding the Sampling/Weld Station with Glycol. The temperature of an MCO in the CSB sampling/weld station is controlled by a cooling system containing propylene glycol. The glycol circulates through the upper shield region that surrounds the upper portion of the MCO. For this scenario, it was assumed that the cooling system failed and the void region along the entire length of the MCO was flooded. The composition of the coolant was assumed to be 100% propylene ($\text{C}_3\text{H}_8\text{O}_2$). More than 12 in. of concrete separates the stations from each other, neutronically isolating them, so only one station is modeled. The personnel access area around the top of the MCO was assumed to be concrete to provide maximum reflection. Calculations were performed for both flooded and drained MCOs and the results are listed in Table 6-33.

Table 6-31. Analysis Results for Multi-Canister Overpack Misloading. (3 sheets)

File ID	Fuel type	Payload cluster		Water density (g/cm ³)		Calculation results ^a		
		Number of baskets	Assemblies per basket	MCO interior	Between storage tubes and outside MCOs	k _{calc}	σ _{calc}	k _{eff}
oc3.1b	Mark IV ^b	Top MCO 1 top 3 middle 1 bottom	Top MCO 0.95 wt% ²³⁵ U scrap 54 Mark IV 0.95 wt% ²³⁵ U scrap	0.0051	0.008	0.4134	0.0018	0.4248
		Bottom MCO 2 top 2 middle 1 bottom	Bottom MCO 0.95 wt% ²³⁵ U scrap 54 Mark IV 0.95 wt% ²³⁵ U scrap					
oc3.1	Mark IV ^b	Top MCO 1 top 3 middle 1 bottom	Top MCO 0.95 wt% ²³⁵ U scrap 54 Mark IV 0.95 wt% ²³⁵ U scrap	0.0051	0.0012	0.3939	0.0015	0.4052
		Bottom MCO 2 top 2 middle 1 bottom	Bottom MCO 0.95 wt% ²³⁵ U scrap 54 Mark IV 0.95 wt% ²³⁵ U scrap					
oc3.2	Mark IV ^b	Top MCO 1 top 3 middle 1 bottom	Top MCO 0.95 wt% ²³⁵ U scrap 54 Mark IV 0.95 wt% ²³⁵ U scrap	1.0	0.0012	0.9248	0.0024	0.9365
		Bottom MCO 2 top 2 middle 1 bottom	Bottom MCO 0.95 wt% ²³⁵ U scrap 54 Mark IV 0.95 wt% ²³⁵ U scrap					

Table 6-31. Analysis Results for Multi-Canister Overpack Misloading. (3 sheets)

File ID	Fuel type	Payload cluster		Water density (g/cm ³)		Calculation results ^a		
		Number of baskets	Assemblies per basket	MCO interior	Between storage tubes and outside MCOs	k _{calc}	σ_{calc}	k _{eff}
oc3.3b	Mark IV ^b	Top MCO	Top MCO	0.0051	0.008	0.4131	0.0017	0.4244
		1 top	0.95 wt% ²³⁵ U scrap					
		3 middle	54 Mark IV					
		1 bottom	0.95 wt% ²³⁵ U scrap					
		Bottom MCO	Bottom MCO					
		1 top	0.95 wt% ²³⁵ U scrap mixed with 1 canister of 1.25 wt% ²³⁵ U scrap					
oc3.3	Mark IV ^b	3 middle	54 Mark IV	0.0051	0.0012	0.3973	0.0019	0.4087
		1 bottom	0.95 wt% ²³⁵ U scrap					
		Top MCO	Top MCO					
		1 top	0.95 wt% ²³⁵ U scrap					
		3 middle	54 Mark IV					
		1 bottom	0.95 wt% ²³⁵ U scrap					
		Bottom MCO	Bottom MCO					
		1 top	0.95 wt% ²³⁵ U scrap mixed with 1 canister of 1.25 wt% ²³⁵ U scrap					
		3 middle	54 Mark IV					
		1 bottom	0.95 wt% ²³⁵ U scrap					

Table 6-31. Analysis Results for Multi-Canister Overpack Misloading (3 sheets)

File ID	Fuel type	Payload cluster		Water density (g/cm ³)		Calculation results ^a		
		Number of baskets	Assemblies per basket	MCO interior	Between storage tubes and outside MCOs	k _{calc}	σ _{calc}	k _{eff}
oc3.4b	Mark IV ^b	Top MCO	Top MCO	1.0	0.008	0.9127	0.0025	0.9244
		1 top	0.95 wt% ²³⁵ U scrap					
		3 middle	54 Mark IV					
		1 bottom	0.95 wt% ²³⁵ U scrap					
		Bottom MCO	Bottom MCO					
		1 top	0.95 wt% ²³⁵ U scrap mixed with 1 canister of 1.25 wt% ²³⁵ U scrap					
		3 middle	54 Mark IV					
		1 bottom	0.95 wt% ²³⁵ U scrap					

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

^ak_{eff} = k_{calc} + 0.0004 + ((0.01)² + (1.645²(σ_c² + 0.002083²))^{1/2}) as defined in Section 6.1. This value should be less than 0.95.

^bInfinite array.

MCO = multi-canister overpack.

Table 6-32. Canister Storage Building Array with Single Storage Tube Containing Two Multi-Canister Overpacks and No Intermediate Impact Absorber.

Case	Description	Calculation results		
		k_{calc}	σ	k_{eff}
csb1.7	10 x 22 x 2 array of dry Mark IA MCOs	0.3417	0.0003	0.3527
csb1.8	10 x 22 x 2 array of dry Mark IA MCOs; one tube contains no intermediate impact absorber	0.3420	0.0004	0.3529
csb1.9	10 x 22 x 2 array of dry Mark IA MCOs with no intermediate impact absorbers	0.34587	0.00032	0.3569
csb4.7	10 x 22 x 2 array of dry Mark IV MCOs	0.3637	0.0004	0.3747
csb14.8	10 x 22 x 2 array of dry Mark IV MCOs; one tube contains no intermediate impact absorber	0.3645	0.0004	0.3754
csb4.9	10 x 22 x 2 array of dry Mark IV MCOs with no intermediate impact absorbers	0.36653	0.00035	0.3775

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

MCO = multi-canister overpack.

Table 6-33. Canister Storage Building Sampling/Weld Station Flooded with Propylene Glycol and Containing a Multi-Canister Overpack.

Case	Description	Calculation results		
		k_{calc}	σ_c	k_{eff}
gly1.2	Mark IA MCO drained	0.27115	0.00058	0.28216
gly1.3	Mark IA MCO flooded	0.89000	0.00103	0.90111
gly4.2	Mark IV MCO drained	0.31053	0.00070	0.32156
gly4.3	Mark IV MCO flooded	0.91457	0.00088	0.92564

Note: The results in this table are for information only. For current results, refer to HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

* $k_{eff} = k_{calc} + 0.0004 + ((0.01)^2 + (1.645)^2(\sigma_c^2 + 0.002083^2))^{1/2}$ as defined in Section 3.2. This value should be less than 0.95.

MCO = multi-canister overpack.

These results show that the cooling system for the sampling/weld station does not compromise criticality safety. The models of the MCO were updated to incorporate the most recent design changes to the MCO internals and the fuel and scrap baskets. In addition, the Mark IA scrap basket was only loaded to its maximum limit of 575 kg. Therefore, the results of these calculations are less than those of similar calculations reported elsewhere in this document.

6.6 CRITICAL BENCHMARK EXPERIMENTS

This section provides justification and shows the validity of the calculational method and neutron cross-section values used in the analyses.

Benchmark experiments are primarily used to confirm two aspects of the neutron transport analysis tool:

- That the computer code has a sound treatment of the neutron transport
- That the nuclear cross section database used in the transport code is in agreement with the relevant integral experiments.

The MCNP computer code (Breismeister 1993) is used worldwide and has been extensively tested with its ENDF/B-V-based cross sections. The code development group at Los Alamos National Laboratory, where MCNP was developed, also has a set of 25 calculational benchmarks that extensively test various options within the code. These 25 benchmarks are used to confirm that new versions of the code give exactly the same answer as before and that executables for

users at other sites give equivalent answers. Hence, the 25 calculational benchmarks supplement additional calculations that are made on experimental benchmarks.

MCNP validation efforts specifically appropriate for low-enriched uranium metal systems have been made that have covered N Reactor fuel elements in water (Wittekind 1991, Wittekind 1992, Wittekind 1993) and low-enriched uranium solutions (Wittekind 1994a). The calculational bias has also been determined (Schmittroth and Ruben 1996), and the methodology of applying the bias and uncertainty is discussed in HNF-SD-SNF-CSER-005 (Kessler 2000). MCNP version 4B was used for the most recent calculation (Ruben and Schwinkendorf 1998, Hillesland and Schwinkendorf 1998, Schwinkendorf 1998, Erickson 1998).

The WIMS-E code (Gubbins et al. 1982) was used in this analysis to illustrate trends while the detailed three-dimensional criticality calculations were performed using MCNP. WIMS-E also has been extensively validated against critical experimental data. Previous validation efforts have covered low-enriched uranium metal billets (Erickson 1992, Schwinkendorf 1985a, Schwinkendorf 1985b), Mark IA fuel assemblies and uranium metal rods (Schwinkendorf 1992a), and low-enriched uranium solutions (Schwinkendorf 1992b, Wittekind 1992).

6.7 SUPPLEMENTAL DATA

6.7.1 WIMS-E Calculations for k-Infinity of Fuel Scrap Loads

Assuming uncontrolled geometry, broken fuel pieces may clump together in such a way as to achieve optimal heterogeneity for self-shielding. If these optimally sized fuel pieces become optimally spaced in water, maximum k_{∞} will result. Previous analysis has shown what the minimum critical masses are for various fuel assemblies, components, scrap, solutions, and uranium billets for a set of ideal geometries, such as sphere or slab (Schwinkendorf 1995). Fuel baskets loaded with scrap are assumed to have their entire volume filled with optimal rods.

The variation in maximum lattice k_{∞} is shown in Figure 6-24 for Mark IV and Mark IA fuel in water. Each point plotted is the maximum value for each particular rod diameter (as the moderator-to-uranium ratio is varied).

In order to understand the degree of conservatism inherent in treating broken fuel pieces as optimal scrap, additional calculations were performed in which the scrap material was assumed to consist of non-optimally sized uranium rods. These rods had a diameter equal to the radial thickness of a fuel assembly's outer element and preserved the cladding in direct proportion to the amount of cladding present in intact fuel. Only one typical rod diameter was varied, and k_{∞} was calculated as the spacing was varied. Figure 6-25 presents these results. The thickness of the uranium in the Mark IV outer element is equal to 0.40275 cm, so this was the diameter of the Mark IV scrap assumed in Figure 6-25. The thickness of the uranium in the Mark IA outer element is 0.3467 cm, so this was the diameter of the Mark IA scrap rods. The maximum k_{∞} for the Mark IV scrap (0.95 wt% ^{235}U enriched) in Figure 6-25 is 1.077. This is more representative

of what broken debris may be and indicates how much conservatism is in the uranium metal optimal rod model (about 15 mk). It is not clear that broken pieces would not clump together in an uncontrolled geometry and thus behave neutronicly as a larger piece of scrap. The optimal scrap model must therefore be used for nuclear criticality safety calculations.

Additional conservatism exists in the assumption of zero fuel exposure. N Reactor fuel stored in the K Basins has documented exposure distributions for each key; these records are backed up by direct measurement performed during the fuel segregation campaign in the early 1980s. Credit could be taken for those keys known to have high burnup.

6.7.2 Rod Versus Spherical Geometry for Scrap or Rubble Model

This section shows that either spheres or cylinders can be used to find the maximum k_{eff} by optimizing the size and spacing of the shape selected. By inference, cylinders may be used to find the maximum k_{eff} for rubble or scrap. That is the applicability of treating a random arrangement of rubble, or scrap, as a pristine lattice of uranium rods in water. The question is whether a regular lattice in cylindrical geometry will necessarily produce a bounding k_{∞} for any arbitrarily shaped chunk of material. The question that was actually checked is whether an explicit spherical lattice optimizes to a higher k_{∞} than a cylindrical one. The comparison between an optimized array of spheres and cylinders was addressed with a series of MCNP calculations comparing a hexagonal pitch rod lattice with an explicit, three-dimensional, lattice unit based on spheres arranged in a face-centered-cubic geometry. The fissionable material assumed was uranium metal with an enrichment equal to 0.95 wt% ^{235}U . For both rods and spheres, a double parameter search was made to find the maximum k_{∞} (as the spacing was varied) as a function of uranium piece diameter. MCNP is not an ideal computer code for applications of this nature. Trends that must be smooth tend not to be because of the statistical nature of the code output. However, WIMS-E does not have a lattice module for spherical geometry.

Theoretically, the two most important parameters in a heterogeneous lattice cell problem are (1) the degree of self-shielding and (2) the neutron spectrum, which depends on the degree of neutron moderation. The degree of self-shielding determines the difference between the flux inside and outside the fuel region. Neutrons are born above the ^{238}U resonances and are thermalized by the water to energies below the resonances allowing them to escape much of the ^{238}U resonance parasitic capture in low-enriched uranium metal systems. Lattice calculations of k_{∞} are greater for optimal heterogeneous systems because the thermalization of neutrons in the water region where there is an absence of strong absorbers (i.e., ^{238}U) increases the resonance escape probability of neutrons returning into the fuel region. The degree of self-shielding is quantified in neutron transport theory using the concept of the mean chord length. The mean chord length, $\langle R \rangle$, can be thought of as the average distance a neutron travels through a material region. A simple formula for calculating the mean chord length (Duderstadt and Hamilton 1976) is $\langle R \rangle = 4 \times (\text{volume/area})$. For a rod, this becomes

$$\begin{aligned}
 \langle R \rangle &= \frac{4(\pi R^2 L)}{2\pi R L} \\
 &= 2R \\
 &= D \text{ for a rod of length } L.
 \end{aligned}$$

For a sphere, this becomes

$$\begin{aligned}
 \langle R \rangle &= \frac{4\left(\frac{4}{3}\pi R^3\right)}{4\pi R^2} \\
 &= \frac{4}{3}R \\
 &= \frac{2}{3}D.
 \end{aligned}$$

If the degree of self-shielding is to be the same between the cylindrical and spherical regions, the mean chord length must be the same. This occurs when the diameter of the cylinder is equal to two-thirds of the diameter of the equivalent sphere, or the sphere diameter is 1.5 times the diameter of the cylinder. Figure 6-26 presents the results of the MCNP calculations. Each MCNP k_{eff} shown in Figure 6-26 was a maximum value, out of 10 MCNP calculations that varied the spacing between either the rods or spheres. The shapes of the two curves show very good agreement with predictions if either the rod or sphere diameter is adjusted to compare results with the chord lengths as described above. It is interesting to note that this maximum k_{∞} is very close to the value that WIMS-E maximized to for uranium rods of the same enrichment (1.09226). The maximum k_{∞} was 1.09082 ± 0.00265 for the cylindrical lattice, and 1.09189 ± 0.00247 for the spherical lattice. Finally, the sphere diameter that produces the maximum is about 1.5 times the diameter that produces a maximum for the rods, as predicted by theory.

The foregoing discussion was intended to solidly establish that whether random, irregularly shaped scrap or rubble is treated as rods or as spheres makes no difference in the calculated value of the maximum k_{eff} of the material. The rod and sphere comparison was in excellent agreement with certain transport theory predictions, and hence greater confidence may be put in the assumption that the primary factors that have significance in determining k_{eff} of the unit lattice are the mean (or effective) chord length and the degree of moderation. These factors are not sensitive to the spatial details of the lattice geometry. Therefore, treating scrap material using parametric calculational results obtained from a cylindrical lattice calculation is valid.

6.7.3 Burnup Effects on N Reactor Mark IV and Mark IA Fuel

All calculations in this chapter were performed using unirradiated fuel. Fuel burnup credit provides an additional safety margin. Experimental data exist for Mark IA fuel showing the impact of fuel burnup on nuclear criticality safety parameters (Kessler 2000).

Burnup calculations have been performed for both Mark IV and Mark IA fuel using hot operating conditions in the N Reactor lattice. Figure 6-27 shows the reactor lattice k_{eff} for both Mark IV and Mark IA fuel. The initial drop in k_{eff} is due to the buildup of equilibrium xenon poisoning. The subsequent behavior of Mark IA fuel shows that k_{eff} decreases monotonically with exposure. Mark IV fuel, because of the lower initial enrichment, shows an increase in k_{eff} caused by the buildup of ^{239}Pu . This behavior has been shown in previous analyses using the HAMMER lattice code (Toffer 1975).

The minimum spherical critical mass for unirradiated Mark IA fuel is 1,893 kg. This calculation assumes optimal moderation (hexagonal lattice spacing of 3.1 in. center-to-center) and full water reflection. With burnup equal to 92 days (which corresponds to 6 wt% ^{240}Pu in total plutonium), this minimum critical mass increases to 2,419 kg, an increase of 28%. With burnup equal to 162 days (9 wt% ^{240}Pu in total plutonium), this critical mass increases to 2,946 kg, an increase of 56% over the unirradiated value. These results are consistent with Toffer (1975), as shown in Figure 11 of the reference.

The variations in maximum lattice k_{∞} and minimum critical mass are plotted as functions of exposure in Figures 6-28 and 6-29. In Figure 6-29, this evolution is plotted as the ratio of the minimum critical mass as exposure increases, divided by the minimum critical mass with no exposure. Burnup effects were treated by using the isotopics generated with the WIMS-E lattice code (Gubbins et al. 1982) for the hot operating lattice environment of N Reactor. These isotopics (except for ^{135}Xe , which was set to zero) were then used in a cold water lattice model where the spacing between fuel assemblies was adjusted to find the maximum k_{∞} and minimum critical mass. The first points plotted in Figures 6-28 and 6-29 are for two days. The point plotted in Figure 6-29 at zero exposure would be exactly equal to 1.0.

The relative shapes of the two curves in Figures 6-27 and 6-28 are quite similar, but there is a significant difference between them. Figure 6-27 shows the change in k_{∞} inside the fixed graphite lattice of the N Reactor using hot operating conditions. These are the k_{∞} values produced by the WIMS-E code as the burnup calculation proceeds. Figure 6-28 shows the maximum k_{∞} when the fuel is removed from the reactor, ^{135}Xe is allowed to decay away, and the fuel is immersed and optimally spaced in a light water moderator.

Results presented in Figures 6-27 through 6-29 were generated with the old 1986 WIMS-E cross section library. Figure 6-30, which illustrates the long-term behavior of the maximum lattice k_{∞} in cold water as the fuel is allowed to decay over a 100-year period, was generated using the new 1994 WIMS-E cross section library, which includes improved treatment of the transuranic burnup chains up through curium. Transuranic isotopic predictions obtained from this new library have been compared to measured N Reactor data (Schwinkendorf et al. 1996). Excellent

agreement with measured data was shown for all transuranic isotopes compared. The results indicate a slight reduction in k_{eff} for both Mark IV and Mark IA fuel as both the fission products and transuranic chains decay. Fission products will eventually decay into other isotopes that provide neutron absorption. Figure 6-31 depicts the Mark IV curve shown in Figure 6-30, with an expanded vertical scale so that the decrease in k_{eff} is more easily seen. This curve is the net result of many unstable or radioactive isotopes changing simultaneously such as fission products, activation products, and transuranic isotopes such as ^{241}Pu and ^{241}Am .

6.7.4 Effect of Uniform Change in Lattice Spacing

A set of MCNP calculations replaced the explicitly defined loading arrangements shown in Figures 6-11 and 6-32 with an infinite lattice of fuel assemblies that was allowed to fill in the available volumes of the fuel and scrap baskets. This allows the non-physical geometry of having partial fuel assemblies "cut off" by the bounding surfaces of the regions into which the fuel lattice is placed. Thus more fuel is added to the baskets than would be loaded in reality. For example, with an infinite lattice, the Mark IV fuel basket holds 60.4 assemblies at a 2.8 in., hexagonal, center-to-center lattice spacing, and the Mark IA baskets hold 56.1 assemblies at the same lattice spacing. Again, this includes partial assemblies that result from extending the lattice all the way out to the bounding surfaces even though they would not otherwise fit. This model is inherently a lower-leakage geometry because fuel pieces exist near the outer boundary that can intercept neutrons that would otherwise have leaked out. The k_{eff} for the 2.8-in. spacing is therefore somewhat higher than the CASE1 base case (by about 15 mk) (see Table 6-5). Figures 6-33 and 6-34 illustrate the trend for the CASE1 and CASE2 geometries (Figure 6-7) when the arrangement of the 48 Mark IA and 54 Mark IV fuel assemblies is replaced with an infinite fuel lattice and the trend as the lattice spacing is increased. Figures 6-33 and 6-34 clearly show that k_{eff} does not peak at slightly higher spacing (the upper and lower scrap baskets in both of these models are driving the k_{eff} of the system). The CASE1 and CASE2 geometries also were modeled with the top and bottom scrap baskets replaced with the intact fuel lattice. The results are shown in Figures 6-35 and 6-36 for full MCO fuel region loadings without scrap baskets on the ends.

6.7.5 Radial Isotopic Profile Evolution During Burnup

Uranium and transuranic isotope buildup in nuclear reactor fuel are functions of both the flux and the neutron spectrum. In a heterogeneous reactor lattice, neutrons born from fission in the fuel escape into the moderator, where they undergo collisions with the hydrogen in the water molecules and lose energy in the process (i.e., they become moderated). For low-enriched uranium metal fuel, a heterogeneous lattice increases k_{eff} by increasing the probability that a neutron will moderate to thermal energies without being absorbed by one of the many resonances in ^{238}U . This probability is increased by allowing the high energy neutrons to thermalize in the absence of resonance absorbers.

Spatial self-shielding in heterogeneous lattices produces spatial flux gradients, and the fluxes have different shapes for different energy ranges throughout the spectrum. The fast, or

high energy, flux has a different spatial shape than the thermal, or lower energy, flux. This shape function drives the isotopic burnup equations differently as the radial position within the fuel piece changes.

In addition, different nuclear reactions occur at different energies. For example, the depletion of ^{235}U occurs when ^{235}U absorbs a neutron and either fissions (80% of the time) or emits a gamma ray. The absorption of neutrons in ^{235}U is much greater for thermal neutron energies because of the "1/v behavior" (proportional to inverse neutron velocity) of the absorption cross section. The depletion reaction of ^{235}U is therefore maximized where the thermal neutron flux is maximized. In contrast, the buildup of ^{239}Pu is an indirect result of the absorption of a neutron in ^{238}U . When ^{238}U absorbs a neutron, it is transmuted into ^{239}U , which emits a beta particle to become ^{239}Np , which emits another beta particle to become ^{239}Pu . Although these beta decays are not a function of the local flux spectra, the neutron absorption in ^{238}U is. Significant neutron absorption in ^{238}U occurs in the epithermal or resonance range of the spectrum. Plutonium production is therefore expected to be greater in those cases where the energy of a partially moderated neutron flux spectrum is higher than that of a fully moderated thermal spectrum.

Figures 6-37, 6-38, and 6-41 illustrate this effect for Mark IV fuel from N Reactor, and Figures 6-39, 6-40, and 6-42 present the Mark IA results. The isotopic burnup profiles inside N Reactor fuel were determined by the WIMS-E code (Gubbins et al. 1982). The regions shown in these figures correspond to the metallic fuel in the tube-in-tube assembly. The non-fuel regions of cladding and coolant at each fuel edge are not shown in the figures. Both the inner and outer elements of the N Reactor Mark IV fuel contained 0.947 wt% enriched uranium metal. The Mark IA fuel used a 0.947 wt% enriched uranium inner and a 1.249 wt% enriched uranium outer tube. The Mark IA fuel, because of its higher k_{eff} , was used as "spike" fuel to reduce radial peaking factors in the reactor.

Hot operating conditions for the N Reactor fuel lattices were obtained from old DCODE (Thierer 1968) results (WIMS-E does not perform heat transport calculations, only reactor physics [Gubbins et al. 1982]). WIMS-E performed the burnup analysis assuming 20 radial subdivisions in the annular fuel regions. The burnup equations were driven locally for each of the radial intervals by the 69-group neutron flux and spectrum for that particular radial interval. Figures 6-37 and 6-39 illustrate the ratio of ^{235}U and ^{239}Pu content, or total fissile, to combined uranium and plutonium versus fuel exposure for Mark IV and Mark IA fuels. The ratios in the bulk central regions of the fuel diminish with exposure, but the ratios at the fuel edges increase with exposure. Figures 6-38 and 6-40 illustrate the ratio of ^{240}Pu content in the plutonium for Mark IV and Mark IA fuels. The ^{240}Pu content, which is a thermal-spectrum neutron poison, also increases with exposure and is at a higher concentration at the fuel edges. Figures 6-41 and 6-42 illustrate the ratio of ^{239}Pu content to combined uranium and plutonium versus fuel exposure for Mark IV and Mark IA fuels. The ratios in the bulk central regions of the fuel and at the fuel edges increase with exposure.

It is observed in Figures 6-41 and 6-42 that the concentration of ^{239}Pu near the surface of the N Reactor fuels is about twice the average concentration of ^{239}Pu in the fuel region.

The upper bound ratio of ^{239}Pu to U is 0.0030 from both figures. The ^{239}Pu content of all plutonium in exposed N Reactor fuel is at least 80 wt%. Therefore, the ratio of Pu to U is less than 0.00375 throughout the fuel region, including the outer edge of both Mark IV and Mark IA fuels. Therefore, the U/Pu atom ratio will always be greater than 267. The inclusion of corrosion from fuels with lower exposure and realistically deeper corrosion penetration in the central region of the fuel will increase this ratio significantly. Uranium and plutonium solutions will always be less reactive than the model based on 1.25 wt% ^{235}U according to discussion of U/Pu ratio in Section 6.4.5.5. Additional discussion also is provided in HNF-SD-SNF-CSER-005 (Kessler 2000).

The enhancement of plutonium production at the outer periphery of nuclear fuel has been documented in the commercial industry (Carlsen and Sah 1981) and predicted by computer codes (Palmer et al. 1982). In addition to radial profiles generated during irradiation because of neutronic effects, thermochemical migration and vapor transport effects have been considered in mixed oxide fuel (Olander 1976). Because of the co-extrusion process of fuels fabrication of Hanford Site production reactor fuel, the fuel is in intimate contact with the cladding (unlike commercial fuel where there is a fuel-cladding gap). Formation of intermetallic phases of plutonium and cladding, near the fuel-cladding interface, have also been suggested as possible concentrating mechanisms for plutonium in Hanford Site production reactors. However, the conclusion has been that formation of intermetallics of this type would not have taken place because of the lower operating temperatures of the fuel. Results reported in this section consider only the neutronics factors driving the evolution of radial plutonium profiles.

The N Reactor burnup results indicate a ^{239}Pu buildup effect near the outer surfaces of the fuel. In all cases, 20 mesh intervals were assumed in the fuel regions. The N Reactor results are likely as pronounced as they will get. These profiles look very much like the profiles in Palmer et al. (1982). Palmer et al. (1982) compared two independent codes (WIMS-E and RADAR) for the calculation of these profiles.

In all burnup calculations reported in this section, the production of plutonium is enhanced at the outer surfaces of the fuel; exposure is increased near the surfaces because the impending thermalized neutron flux is not depleted by self-shielding as it is in the central region of the fuel. However, this effect is at least partially mitigated by the fact that enhanced fissile uranium depletion also occurs near the fuel surfaces. In addition, the increased plutonium production near the surface also has a higher ^{240}Pu content. The overall effect on k_{eff} as discussed above and in Section 6.4.5.5, is that sludge modeled as 1.25 wt% ^{235}U is more reactive than the compositions containing plutonium isotopes near the fuel edge.

Figure 6-1. Fuel Assemblies Storage Basket.

2G96100105.5
SNFP

Figure 6-2. Fuel Scrap Storage Basket.

2G06100105.4
SNFP

Figure 6-3. Multi-Canister Overpack and Storage Tube.

Note: The lower impact absorber is 30 in. tall and the middle impact absorber is 18 in. tall.
This figure is of a preliminary design and not to scale.

Figure 6-4. Multi-Canister Overpacks in Vault Tubes (Finite Array).

2G97020143.2

Figure 6-5. Multi-Canister Overpack in the Multi-Canister Overpack Handling Machine.

Figure 6-6. Model C, Multi-Canister Overpacks in Vault Tubes (Infinite Lattice Cell) and Model D, One Multi-Canister Overpack Encased in Concrete.

MODEL "D" ONE MCO ENCASED IN CONCRETE (Hypothetical geometry)

Figure 6-7. Input Models for Normal Case.

Figure 6-8. Multi-Canister Overpack Loaded with Single-Pass Reactor Fuel, using Mark IA Scrap Baskets. Lattice Spacing Optimized using WIMS-E.

Figure 6-9. Typical Process Bay at the Cold Vacuum Drying Facility.

Figure 6-10. k_{eff} Versus Water Level Internal to a Mark IV Loaded Multi-Canister Overpack.

Figure 6-11. Loading Arrangement for Fuel in Multi-Canister Overpack to Produce Maximum K-Effective.

H9909073.1R7
SNFP

H9909073.3R4
SNFP

Figure 6-12. Reduced Loading for Mark IA Fuel in Multi-Canister Overpack (Twelve Assemblies Removed per Basket).

Figure 6-13. k_{eff} Versus Lattice Spacing - Mark IA Fuel Assemblies.

Note: The fuel assembly array first contacts the basket side at a lattice spacing of 2.8041 in. The k_{eff} for this spacing is 0.82268 with an uncertainty of 0.00264 and for a spacing of 2.80 in. when the k_{eff} is 0.81923 with an uncertainty of 0.00342.

Figure 6-14. MCNP Mark IA Fuel Basket Grid.

```

fuel basket containing intact fuel assemblies
2      0      -17      18      -19      20      -21      22      u=8      lat=2
      fill=-5:5      -5:5      0:0
      2      2      2      2      2      2      2      2      2      2      2      $ 1
      2      2      2      2      2      2      3      3      3      2      2      $ 2
      2      2      2      2      3      3      3      3      3      3      2      $ 3
      2      2      2      3      3      3      3      3      3      3      2      $ 4
      2      2      3      3      3      2      2      3      3      3      2      $ 5
      2      2      3      3      2      2      2      3      3      2      2      $ 6
      2      3      3      3      2      2      3      3      3      2      2      $ 7
      2      3      3      3      3      3      3      3      2      2      2      $ 8
      2      3      3      3      3      3      3      2      2      2      2      $ 9
      2      2      3      3      3      2      2      2      2      2      2      $ 10
      2      2      2      2      2      2      2      2      2      2      2      $ 11
      1      2      3      4      5      6      7      8      9      10      11

```

Figure 6-15. Mark IA-Loaded Multi-Canister Overpack with Half of the Inner Elements Near the Inner and Outer Fuel Region Annulus Edges.

Figure 6-16. Mark IA-Loaded Multi-Canister Overpack with Half of the Inner Elements Loaded Near the Center of the Fuel Region Annulus.

Figure 6-17. Lattice k -Infinity and Temperature Coefficients for N Reactor Fuel Assemblies.

Figure 6-18. Drop Analysis Input Models.

Figure 6-19. Water-Flooded, Mark IV-Loaded Multi-Canister Overpack Being Lowered into a Canister Storage Building Storage Tube.

Figure 6-20. Interspersed Moderation for Dry Mark IV Multi-Canister Overpacks in the Canister Storage Building.

Figure 6-21. Interspersed Moderation for Flooded Mark IV Multi-Canister Overpacks in the Canister Storage Building.

Figure 6-22. Maximum k-Infinity Versus Rod Outside Diameter, Including Cladding.

Figure 6-23. k-Infinity Versus Spacing Smaller, Non-optimally Sized Rods.

Figure 6-24. Interspersed Moderation, Water Density Variation inside Multi-Canister Overpack.

Figure 6-25. Interspersed Moderation, Water Density Variation outside Multi-Canister Overpack.

Figure 6-26. Maximum k-Infinity Versus Outside Diameter, Face-Centered-Cubic Spherical versus Hexagonal Rod Lattices.

Figure 6-27. k -Infinity Versus Exposure for N Reactor Hot Operating Conditions.

Figure 6-28. Maximum Lattice k -Infinity Versus Exposure for N Reactor Fuel in Cold Water.

Figure 6-29. Minimum Critical Mass Ratio Versus Exposure for N Reactor Fuel in Cold Water.

Figure 6-30. Maximum Lattice k-Infinity Versus 100-Year Decay for N Reactor Fuel.

Figure 6-31. Maximum Lattice k-Infinity Versus One Hundred-Year Decay for N Reactor Mark IV Fuel.

Figure 6-32. Loading Arrangement for Mark IV Fuel in Multi-Canister Overpack.

Figure 6-33. Reduced Intact Fuel Loading with Scrap Baskets — Mark IA Assemblies.

Figure 6-34. Reduced Intact Fuel Loading with Scrap Baskets — Mark IV Assemblies.

Figure 6-35. Reduced Intact Fuel Loading — Multi-Canister Overpacks Loaded with Mark IA Assemblies.

Figure 6-36. Reduced Intact Fuel Loading — Multi-Canister Overpacks Loaded with Mark IV Assemblies.

Figure 6-37. Evolution of the $^{235}\text{U} + ^{239}\text{Pu}$ Radial Profile for Mark IV N Reactor Fuel.

Figure 6-38. Evolution of the wt% ^{240}Pu Radial Profile for Mark IV N Reactor Fuel.

Figure 6-39. Evolution of the $^{235}\text{U} + ^{239}\text{Pu}$ Radial Profile for Mark IA N Reactor Fuel.

Figure 6-40. Evolution of the wt% ^{240}Pu Radial Profile for Mark IA N Reactor Fuel.

Figure 6-41. Evolution of the ^{239}Pu Radial Profile for N Reactor Mark IV Fuel.

Figure 6-42. Evolution of the ^{239}Pu Radial Profile for N Reactor Mark IA Fuel.

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7.0 CONFINEMENT

The multi-canister overpack (MCO) represents the first confinement boundary for the spent nuclear fuel (SNF) during storage. The MCO is a single-boundary system that serves the following uses during its lifetime:

- The innermost SNF container as part of the transport shipping packaging
- The process vessel for the SNF during cold vacuum drying
- The container for the SNF during process validation and interim storage at the Canister Storage Building (CSB).

7.1 CONFINEMENT BOUNDARY

The confinement boundary for the MCO is composed of the MCO lower end cap, the MCO shell wall, collar, and the MCO shield plug assembly, including the metallic seal (Figure 7-1).

- The lower end of the MCO is a machined cap made of 304L stainless steel. A depression at the bottom of the cap collects bulk water for removal from the MCO. The minimum thickness for the lower end cap is 0.5 in. where it is welded to the shell wall.
- The shell wall of the MCO is a 24-in.-outside-diameter pipe section with a 0.5-in.-thick wall section. The shell is made of 304L stainless steel. The shell wall, collar, and lower end of the MCO will be manufactured and pressure and leak tested together with the shield plug assembly at the fabricator's site.
- The shield plug assembly includes the plug body itself and the penetrations and devices necessary for interface to operating and safety equipment. The devices at the ends of the penetrations from the interior of the MCO form a portion of the confinement boundary for the MCO.

The MCO confinement and pressure boundary is designed and manufactured in accordance with the standards in the *Boiler and Pressure Vessel Code* (ASME 1998), Section III, Subsection NB. The MCO design pressure is 450 lb/in² gauge at 132 °C (270 °F) when the cover cap is attached and 150 lb/in² gauge when the cover cap is not attached. The MCO maximum design temperature after fuel is inserted into the MCO is 132 °C (270 °F). Additional details are provided in Sections 2.2.6.2.1 and 2.2.6.2.2, and in Chapters 4.0 and 12.0.

7.1.1 Confinement Passages

Passages are bored through the shield plug to the interior of the MCO. The passages terminate on the outside of the shield plug as ports, with devices and fittings that allow the MCO to interface with operating and safety-related devices. Three devices are associated with the ports: two process valves and one pressure relief device. Fittings and covers that go over the devices offer additional confinement for the MCO. A fourth port is available if needed but will be plugged during normal conditions.

The two process valves on the external face of the MCO are connected to process equipment. These process connections are used to introduce fluids into and remove fluids from the MCO. The long process tube path (port 3) has a bore of 0.59 in.; the path connecting to the internal high-efficiency particulate air (HEPA) filters (port 2) has a nominal diameter of 1 in. The process path from port 3 is protected by a wire screen (2-mm mesh openings) on the interior of the MCO to prevent larger particles from being moved out of the MCO (Figure 1-9). This defines confinement as the retention within the MCO of all materials greater than 2 mm in diameter.

A bank of four HEPA filters welded to the underside of the shield plug filters exit gases during vacuum drying through port 2. The HEPA filters, which are not testable after the MCO is assembled, are connected to two penetrations (ports 1 and 2) in the MCO shield plug. These HEPA filters serve to minimize the potential for escape of contamination and reduce the need for external MCO filtration.

7.1.2 Seals, Welds, and Closure

MCO confinement is established by a combination of seals and welds. The shell wall pipe, collar, and MCO bottom are joined by welding, and the joints are inspected and examined before the MCO is loaded with fuel. The shield plug penetrations are sealed during transport to the CSB. After receipt at the CSB, a single cover cap is placed over the entire shield plug and welded to the MCO shell wall collar at the CSB.

The shield plug is secured to the MCO at the K Basins using a mechanical closure. A threaded locking ring is placed into the MCO collar neck extension after the shield plug is inserted. Once assembled, 18 set screws in the locking ring are tightened down into the shield plug to push the shield plug into the seal between the MCO shell and the shield plug. Information on the bolt torquing requirements is provided in Appendix 4 of the MCO Design Report (Goldmann 2000a). The locking and lifting ring is installed before the cask-MCO is transported to the Cold Vacuum Drying Facility (CVDF). The seal is maintained during thermal transients and drop accidents by developing and maintaining sufficient preload through the seal. This type of seal system was qualified during the prototypic testing program and by analysis, and as a result, only a routine examination of the main seal will be performed at the K Basins.

The shield plug is installed in the MCO neck underwater in the load-out pit. Installing the locking and lifting ring and tightening the 18 set screws are accomplished in the load-out pit with the immersion pail system in the "up" position. Subsequently, the main seal is leakage-rate tested at the CVDF.

7.2 REQUIREMENTS FOR NORMAL CONDITIONS OF STORAGE

The MCO confinement boundary is tested at various locations to ensure the leakage rate requirements are met (Goldmann 2000b). When mechanically sealed, the maximum acceptable leakage rate is 1.0×10^{-5} standard $\text{cm}^3/\text{s}\cdot\text{atm}$. When in a welded sealed condition, the maximum acceptable leakage rate is 1.0×10^{-7} standard $\text{cm}^3/\text{s}\cdot\text{atm}$. The MCO collar, shell, bottom end, shield plug assembly, and individual cover caps are leakage rate tested at the fabricator's facility before the subassembly is loaded with fuel. The mechanical closure system will be qualified for suitability and only a routine examination of the seal occurs at the K Basins after fuel loading and MCO closing are complete. The mechanical seal of the MCO is leakage rate tested at the CVDF to determine acceptability for storage at CSB. It is not anticipated that the 18 set screws in the mechanical closure will need retorquing while the MCO is in storage at the CSB. This is based on test results after the prototypic MCO had gone through five thermal cycles and vibration testing. The mechanical seal maintained leaktightness after testing of the prototype was complete. After receipt at the CSB, a cover cap is welded to the MCO shell over the top of the shield plug and locking and lifting ring. The cover cap weld is capable of being volumetrically examined by ultrasonic inspection equipment once the MCO is sealed and prepared for interim storage in the CSB storage tubes.

During sealed interim storage in the CSB, the MCO has no penetrations that are open for sensing or venting. The atmosphere inside the MCO is not interacting with the atmosphere of the CSB storage tube.

The MCO is stored in the CSB storage tube. This tube is fitted with a shielded plug. The operating deck of the CSB is surrounded by the above-ground portion of the CSB's walls and by the roof. Thus, under long-term storage, confinement is provided by the MCO and by the storage tube. The CSB is designed to withstand the loads imposed by Hanford Site hazards (e.g., tornadoes, seismic events [see Chapter 2.0]) as well as to mitigate or prevent accidents such as MCO drops into the storage tubes (see Chapter 3.0).

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Figure 7-1. Schematic Diagram of Multi-Canister Overpack Confinement Boundaries.

Functions and Legend

Port number	Function		
1	Filtered backup port	⊗	Appliance (5) (potential confinement boundary)
2	Filtered process exit port	—	Cover plate position (5) (potential confinement boundary)
3	Long axial process tube	—	Confinement boundary (either shield plug or cover cap claimed at one time)
4	Safety-class rupture disk and backup process port		

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8.0 MULTI-CANISTER OVERPACK CONFIGURATION

The multi-canister overpack (MCO) is subjected to different environments and operations from the time it is transported to the K Basins until it is placed in interim storage in the Canister Storage Building (CSB). Chapters 3.0 and 4.0 provide detailed evaluations of the MCO for these different environments and operations. Specific details of the activities that occur in each of the Spent Nuclear Fuel (SNF) Project facilities are presented in the respective safety analysis reports (SARs). This chapter briefly summarizes the processing path of the MCO and describes the configuration at each stage of this path. A better understanding of the MCO process steps and the conditions that the MCO is subject to will facilitate application of the limits presented in Chapter 12.0 in development of the necessary technical safety requirements and procedures and the maintenance and operating procedures needed for the various processing steps.

8.1 K BASIN FUEL REMOVAL

The MCO shell is loaded into an empty transport cask at the CSB. A protective cover prevents the interior of the MCO from being contaminated with foreign materials. The cask-MCO is unloaded from the conveyance and placed into an immersion pail that is located in the south load-out pit at the K Basins. The cask lid is then removed, followed by the removal of the locking ring. The pail lid is then installed and pail is lowered.

The K Basin Fuel Retrieval System will be used to retrieve SNF and package it into MCO baskets in preparation for loading into the MCO. SNF will be moved, using the K Basin monorail cranes, from the basin storage locations to a storage area near the fuel retrieval equipment in the west bay of the basin. Fuel canisters will be loaded into the fuel retrieval system's primary cleaning machine for cleaning and for loosening stuck SNF scrap. Following cleaning, the canisters will be discarded and some portion of the SNF will be visually inspected for removal of debris and sludge.

SNF scrap will be loaded into one of two types of MCO scrap baskets. Intact fuel elements will be loaded into one of two types of storage baskets. Operational controls will ensure the fuel is loaded into the correct MCO basket type and fuel loading limits are not exceeded. Proper loading protocols will be established at K Basins. Filled baskets will be moved to the MCO basket queue to await preparation for loading into the MCO. The MCO basket queue can hold 10 MCO baskets.

MCO baskets are moved manually from the loading queue to a shuttle cart. The shuttle moves the basket through the transfer channel to a position next to the immersion pail. A loading funnel is placed into the neck of the MCO to prevent damage to the MCO closure surfaces during loading of the SNF baskets (Praga 1998b). This step may be performed before the cask-MCO is put into, and flooded in, the load-out pit. The MCO basket is inserted into a cask-MCO using a gantry and telescoping mast. A rail structure is provided to support the gantry. After the correct number of baskets has been loaded, the loading funnel is removed from the neck of the MCO.

The shield plug assembly is prepared for installation while the MCO is underwater in the load-out pit. Assembly includes attachment of a shield plug lift fixture and installation of the long, axial process tube. The shield plug lift fixture is attached to the auxiliary hoist using the shield plug lift rig and moved over the submerged MCO containing the SNF payload. After the sealing surface on the MCO collar is cleaned underwater, the long process tube is inserted into the top basket's center pipe, and the shield plug assembly is slowly lowered down and into the MCO. Once shield plug insertion is complete, the cask-MCO assembly is raised up out of the water, and the lift fixture is decontaminated and removed from the shield plug. Any additional decontamination steps may occur at this time.

The threaded locking ring is installed in the MCO and the set screws tightened. The shield plug valve alignment allows the MCO to "breathe" into the cask cavity after the cask lid is installed. Leak testing at this point is not necessary because the mechanical closure system will have been pressure tested and qualified at the fabricator's plant. The cask lid is installed before the cask is inerted and lifted from the pit. Then the cask-MCO assembly is inerted, put on the conveyance, and secured. The cask-MCO weighs approximately 30 tons when fully loaded with SNF (see Section 7.2).

After assembly and tightening of the cask lid, the cask and MCO diving bell chamber above the water levels will be reduced to 1.6 vol% oxygen in the contiguous volume. The reduction of oxygen from the inside of the cask body will be accomplished by gas buildup and subsequent release of the mixed gas from the cask. The cycle will be repeated until the desired dilution inside the cask body is achieved. The dilution method is described in HNF-2833, *Inert Gas Requirements for Cask Loading* (Pajunen 1998b). Although the report specifically mentions helium as the dilution gas, nitrogen is also being evaluated. A revision of HNF-SD-SNF-FRD-025, *Functions and Requirements for K Basins Facility Upgrades for Transport* (Ard 1998), is in the planning stages and will include the "Helium Purge System" for the cask-MCO assembly in the functions and requirements document for each basin's load-out system. The annual updates of this Topical Report will conclude this discussion.

8.2 COLD VACUUM DRYING FACILITY

8.2.1 General Facility Description

The Cold Vacuum Drying Facility (CVDF) will receive loaded MCOs from the K Basins. The CVDF is a new, stand-alone, modular structure located in the Hanford Site 100 K Area. The site selected for the CVDF is west of the 165/190 KW Building. This site is close to all required utilities and within the inner security boundary. The new facility is located near the path the fuel transport truck will take leaving the 105 KW Reactor Building.

The CVDF contains five process bays (including one empty bay) within a single-story, pre-engineered metal frame and concrete panel building containing a second-level mezzanine. Each process bay is designed as an enclosure for an MCO cask transporter, without the tractor

attached, and includes operational space necessary to meet the functional requirements of the CVDF. Process bay construction is designed to provide separation and confinement within each bay. Each process bay area has a width of 30 ft and a nominal length of 60 ft. The height of the process bay is nominally 32 ft, which is dictated by the manned access required at the working level of the shipping cask, the crane access required to remove the cask lid, and the physical requirements for all the operating equipment.

Each process bay provides ground space for the following items:

- A cask transporter, without the tractor attached (a safety-related confinement zone)
- Personnel circulation and functional space around the cask transporter
- Seismic restraint hold-down devices for the cask-MCO, if required
- Vacuum drying system equipment and pump assemblies
- Access to the working level of the cask and transporter
- Radiological control between the process bay and the access corridor (building design allows for anti-C clothing installation and removal and personnel surveys when moving between rooms)
- Bridge crane access for removal of the cask lid and maintenance on equipment
- Auxiliary services, including inert gas, and pneumatic and electrical power
- A cabinet for supplies.

Access to the working level of the cask is accomplished using a mezzanine level with space for the following items:

- Connections from the vacuum drying system to the cask-MCO
- Heating, ventilating, and air conditioning equipment and electrical panels
- Process hood assembly and handling means
- Leakage rate testing equipment.

8.2.2 Cold Vacuum Drying

The sequence of actions that occur within the CVDF between the time that an MCO is ready for shipping from the K Basins until the process bay is ready to receive another MCO can be summarized as follows.

1. **Receiving Activities.** K East or K West Basin operations will be notified that CVDF operations is ready to receive a shipment. CVDF operations will select a bay and notify the truck driver, who then positions the transporter in front of the corresponding door. The bay door is opened and an exhaustor trunk is attached to the truck's exhaust pipe so that diesel fumes do not enter the CVDF. The truck backs into the bay and locates the transporter.
2. **Preparation Activities.** Once the MCO has been accepted, the trailer is secured, and the tractor is disconnected and driven out of the facility.
3. **Receiving Inspection and Acceptance Activities.** Receiving inspection activities will be performed to comply with special nuclear material accountability requirements, confirm smearable contamination levels on the cask are within transportation and facility limits, provide dose rate data for the cask and MCO to support planning or radiological work, and preserve worker exposure as low as reasonably achievable (ALARA). When the MCO is received at the CVDF, the cask is inspected in accordance with the facility's requirements.
4. **System Connection.** After the MCO assembly is secured and determined to be within acceptable limits for smearable contamination, radiation monitoring instrumentation will be activated; water hoses attached to the cask ports; the cask lid removed and replaced with a cask process lid; the bridge crane carrying the process hood and seal ring and the process connection spools put in place; then the process hoses attached to the MCO top ports.
5. **Cold Vacuum Drying Activities.** Cold vacuum drying activities include heating the MCO to the drying temperature (the water in the annulus between the MCO and the cask is left in order to obtain effective heat transfer to the materials inside); draining the water from the MCO; executing a sequence of purge and evacuation cycles; performing an extended vacuum drying cycle to ensure free water removal; backfilling with helium; and ultimately, cooling the cask. All these operations are run from the control room. In addition, there is also a demineralized water flush of the process water conditioning line as part of the drain process.
6. **Postprocessing Examination.** Leakage rate testing is performed on the MCO mechanical seal before the MCO leaves the CVDF.
7. **System Disconnection.** System disconnection activities are the reverse of the preparation activities. The process hoses are disconnected; they are to be handled as

contaminated items (ends are stored in the appropriate holders). The process hood is removed. The water in the annular space between the cask and the MCO is drained. The cask top is placed, sealed, and the annular space dried with instrument air. Instruments are removed from the transporter. The tractor arrives and is connected to the transporter. The transporter is released and driven away.

8.2.2.1 Vacuum and Purge System. The function of the vacuum and purge system is to remove the bulk water from the MCO and to dry the SNF under vacuum using a cycle of evacuation and inert gas backfill and purge stages. There will be one vacuum and purge system for each of the process bays. The final product will be a dried MCO that has been backfilled with inert gas. The vacuum and purge system will be required to interface with the monitoring and control system to allow for computer control of the MCO during all stages of the drying operation.

8.2.2.2 Tempered Water Control System. The MCO temperature control system maintains the MCO at the proper operating temperatures during all stages of the MCO drying operation. This includes heating the MCO during vacuum drying and cooling the MCO after vacuum drying. The temperature control system pumps water at various design temperatures through the annular space between the MCO and the cask at a rate of 20 gal/min and a design pressure of 20 lb/in². During the vacuum drying phase of the operation, the temperature control system maintains a continuous operating temperature for a period of about 2 days per drying operation. After vacuum drying, the temperature control system cools the MCO and cask down to an acceptable temperature for shipping. The temperature control system interfaces with the monitoring and control system to allow for computer control of the MCO during all stages of the cold vacuum drying operation.

8.2.2.3 Monitoring and Control System. The monitoring and control system is designed as a fully integrated control system that provides not only process control but data acquisition and management. The system uses digital signaling between a distributed network of programmable logic controllers and driver software, which can be handled by personal computer-sized hardware. Each local personal computer has a view screen with dynamic graphical display to show the change in the operating parameter. Each personal computer control station in the control room allows total access to all systems so that redundancy is achieved if a computer becomes inoperative. Access to the level of control can be programmed into the control system software so that only authorized personnel may operate the system. A television camera located in each process bay allows control room personnel to view the operation on a television screen near each personal computer control station.

8.3 CANISTER STORAGE BUILDING

The CSB is located in the 200 East Area of the Hanford Site. The CSB is a hazard category 2 facility that consists of three equally sized below-grade concrete vaults, with total approximate dimensions of 180 ft x 165 ft x 48 ft deep, covered by a concrete operating deck. An above-grade steel operating area structure is located above the vaults. Support functions and equipment are housed in a smaller building at the north side of the operations building.

The reference elevation of the facilities at the top of the operating deck is 709 ft above mean sea level. The basemat is nominally 5 ft, 6 in. thick; its surface elevation is 666 ft, 9 in. The distance from the surface of the basemat to the underside of the operating deck is 37 ft, 3 in. The exterior walls and air inlet and outlet plenums are 4 ft, 6 in. thick. Interior partition walls between vaults are 3 ft thick.

A 5-ft-thick, standard, reinforced concrete operating deck structure forms the at-grade portion of the CSB. The operating deck and other operations areas are enclosed in a steel building. The operating deck contains numerous through-thickness steel sleeves (floor embeds) that receive the storage tubes and floor plugs for both the MCO and the MCO overpack locations in vault 1. Each storage tube will contain two impact absorbers, one at the bottom and one between MCOs, to mitigate the consequences of a dropped MCO. The impact absorbers are designed to prevent the breach of the storage tube and the MCO.

MCOs are transported from the CVDF in shielded casks and received into the truck vestibule at the northwest corner of the CSB. The cask containing the MCO is unloaded by a gantry-type receiving crane and transferred to a below-grade service station at the north end of the operating area. In the service station, the cask lid is removed and the MCO is removed from the cask by the MCO handling machine (MHM). The MHM is an air-atmosphere hoist that provides radiation shielding and is equipped with a ventilation system and a manual control system.

MCOs are seal-welded at the CSB sampling/weld stations. These stations are used to perform sampling of the MCOs as well. The actual procedures and forms used to perform the following activities have yet to be prepared. A checklist or similar form will be used for all the sealing and welding activities to record the equipment, process parameters, components used, operators, and similar information for the file. Existing CSB facilities, shielded holders, hoods, ventilation systems, hoists, and utilities will be used to accomplish the sealing. The welding, leakage-rate testing, dye-penetrant examination, possible ultrasonic testing, and possible monitoring activities, along with the associated personnel, tooling, fixtures, and materials to do the tasks, are also provided. The cover cap closure welds are made in accordance with the *ASME Boiler and Pressure Vessel Code*, Section III, subsection NB (ASME 1998), and they are examined in accordance with the *ASME Boiler and Pressure Vessel Code* and appropriate code cases.

MCOs are transferred via the MHM to the weld station for cover cap welding and weld seal leak testing. Welding, weld inspection, and leak testing are performed in a shielded receptacle within the sampling/weld station pits by hands-on operations and miscellaneous support equipment. Although not anticipated, the capability exists for any needed adjustments to the interior gas of the MCO to be made before the cover cap is welded in place.

The MCO cover cap is brought in and positioned on top of the prepared MCO and its rotational orientation is verified. The welding head is positioned on top of the cover cap, and the head is physically adjusted, feed wire checked, and needed dimensions and positions verified.

The locational index test wrench is removed from the cover cap penetration, the weld backer gas purge tool is inserted into the penetration, and the gas flow rate is adjusted.

The welding machine power supply and controls are activated, and the welding program is verified. The first weld pass (root) is made and the pass is dye-penetrant examined. The remaining passes are performed up to mid-weld level, and a second dye penetrant examination is performed. The remaining passes are laid down to complete the weld, and the whole weld is leakage rate tested. After successful leakage rate testing, the exposed portions of the last weld passes are dye-penetrant examined.

The test plug is tightened in the cover cap penetration, and the plug is leakage rate tested. Upon successful plug leakage rate testing, the penetration cover plate is positioned in the penetration socket, and it is manually welded, with dye-penetrant examinations performed on the root and final passes.

After helium leak testing, welded MCOs are transferred from the weld station to the storage tubes for interim storage. Some periodic monitoring of the sealed MCOs is expected during interim storage; the only other operations would be conducted in response to off-normal events.

When the cover cap is on the MCO, the penetration has a number of potential uses:

- Putting weld backer gas under the cover cap during welding attachment to the collar to reduce oxidation of the molten weld metals and surrounding heat-affected areas
- Putting helium under the cover cap so the attachment weld may be leakage-rate tested with a mass spectrometer from the outside
- Leakage-rate testing the MCO Helicoflex main seal
- Hydrostatic testing of the cover cap after installation of the cover cap on the collar
- Allowing access to port 2 in the shield plug for gas sampling, pressure adjustment, and/or exchange inside the MCO.

8.4 MULTI-CANISTER OVERPACK FACILITY-SPECIFIC CONFIGURATION

The previous sections have briefly described the functions of the various facilities the MCO will pass through or reside in. The information that follows describes the configuration of the MCO at each of the various steps in its process cycle. The description for each step shows (1) what is installed on the MCO, (2) how the MCO is configured, (3) what components are active on the MCO, and (4) what the possible uses are for the MCO in a particular configuration. This discussion centers on the shield plug's four ports and how they are used during the MCO process cycle. The steps are presented sequentially so that changes to the configuration may be

easily identified. This allows operations to plan for these changes and will assist in determining the equipment, features, support, and procedures that are needed at each facility. Maintenance needs also are discussed to the extent that they have been identified.

A cross-sectional view of the shield plug and a top view of the ports in the shield plug are provided in Figure 1-1. Port numbers are identified on the figure, and in this description, they will be referred to by number. To begin the description, the internal components of the MCO will be discussed first.

8.4.1 Multi-Canister Overpack Internal Components

The internal components are the internal filters, long process tube, and safety-class rupture disk inlet. The fuel baskets are discussed in Section 1.2.2. The internal filters and the inlet for the safety-class rupture disk are protected at the bottom of the shield plug by a guard plate. The guard plate and side structure create an armored chamber where the internal filters and inlet for the rupture disk are protected from damage during drop accidents. Gas flows from the MCO proper into the chamber through eight screened (2-mm mesh), 0.875-in.-diameter holes in the guard plate. The guard plate and supporting side structure also are shown on Figure 1-1.

8.4.1.1 Internal Filters. The internal filters are made of metal high-efficiency particulate air (HEPA) filter media. The filter bank is made up of four filter elements that are in an all-welded assembly affixed to the inside of the shield plug face. Individual filters are each approximately 2.6 in. in diameter and 10 in. long. Each filter has about 1.5 ft² of pleated filter medium for a total of 6 ft² for the assembly. These filters are specified to maintain at least half of their flow capacity after design basis accident (DBA) drops, but the HEPA filtering capability of the metal filter media may be compromised during DBA drop events.

8.4.1.2 Long Process Tube. This tube is a 1-in. schedule XXS stainless steel pipe, which is used for bulk water removal prior to cold vacuum drying and for the introduction of process gases during cold vacuum drying. The tube may also be used to reflood the MCO in the event it has to be refilled with water once it has been drained. The bore of this pipe is approximately 0.59 in. and runs to the pocket at the bottom to the MCO. The pipe inserts into a cavity that is screened to 2 mm to prevent particles larger than 2 mm from leaving the MCO.

8.4.1.3 Safety-Class Rupture Disk Inlet. The safety-class rupture disk protects the MCO pressure boundary. This inlet path is also available for use by the vacuum drying system as a backup process exit path in the event the MCO internal filters become blinded or unusable for some reason.

8.4.2 Multi-Canister Overpack Ports

As noted, the ports play a critical role in determining the MCO configuration. Each port is capable of being variably equipped with a cover that can (1) be used to seal the port during the

handling, transportation, and storage steps, (2) afford physical protection to the appliance underneath the cover inside the port, and (3) include an orifice for reliably regulating gas flow from the MCO as needed. All three features will not be used on all ports, and all features will not be used at the same time on a single port.

The ports are set up to be used as process ports. A mechanical operator is fastened to the port (the cover has been removed prior to the installation of the operator) and operates the integral valve. Under normal conditions, the valve is never removed from the shield plug. A detachable mechanical operator will be used at each facility to open and close the valves, to perform maintenance, or change the appliances.

The mechanical operator is an integral part of the facility piping system and is attached to the MCO before the valve in the MCO shield plug is operated. The valve can be used to slowly "bleed down" the MCO in the event that internal pressure has built up inside the MCO. After bleed-down is complete, the valve is opened further to accommodate gas flows and to support any needed interior processing of the MCO.

The external features and internal components of the MCO are connected with four passageways leading through the shield plug. These passages have several bends in them to prevent direct unattenuated radiation from leaving the interior of the MCO.

8.4.2.1 Port 1. Port 1 uses a 1.0-in.-diameter passage leading to the internal HEPA filter bank. This port will likely be plugged with a removable threaded plug and not used. However, this port is a multipurpose port and could serve the following pressure management needs when configured to suit.

- This port could serve as the process pressure relief of the MCO in the unlikely event that excessive internal gas production be discovered. This filtered relief path allows the excess gases to leave the MCO at a pressure that is above surrounding atmospheric pressure but is still well below the safety-class pressure relief rupture disk setting for the MCO. When configured to address this hypothetical excessive gas production, this port would have a process pressure relief valve installed at the port. This valve will reseal once the MCO internal pressure has been bled down to the reset pressure for the process pressure relief valve. This mode of operation for the MCO would be likely if staging of the MCOs at the CSB is required and allows the MCO to communicate with the surrounding atmosphere, effectively in one direction only (i.e., out of the MCO).
- This port also may serve as a receptacle for a process rupture disk to relieve pressure from the inside of the MCO. After process conditioning, a rupture disk may be installed in this port to relieve the pressure. Once actuated this rupture disk allows the MCO to communicate in both directions with its surrounding atmosphere. The relief setting for this process rupture disk, if used, would be at a pressure well below the actuation setting for the safety-class rupture disk.

- As a possible future option, this port also may serve as a receptacle for an external HEPA filter holder. This port, when configured with an external HEPA filter, allows the MCO to be staged or stored at roughly CSB storage tube pressure. Using this option allows the MCO to communicate both ways with the surrounding atmosphere in the tube.
- This port may have the potential for being accessed when a cap is welded on the MCO for long-term storage or used at the CSB for MCO gas sampling. It is likely that the port will be sealed with a cover plate, but an appliance may also be installed.

8.4.2.2 Port 2. Port 2 uses a 1.0-in.-diameter passage (same inner diameter as schedule XXS 1-in. pipe) leading to the internal HEPA filter bank. This port is used for process purposes and is set up to allow the use of the operator to open and close the integral valve. This port also allows gases exiting the MCO during the cold vacuum drying step to be internally filtered to keep as much of the radioactive particulate materials inside the MCO as possible.

8.4.2.3 Port 3. Port 3 is a process port with the integral MCO valve and an approximately 0.59-in.-diameter passage leading to the long axial process tube. This port is used for (1) the removal of bulk water at the CVDF, (2) the introduction of inert gases into the MCO, and (3) the introduction of process mixtures needed for cold vacuum drying. Reflooding of an MCO with water may be accomplished through this port. The bolt pattern used to affix the valve operator to this port is different from the other ports to preclude an incorrect valve operator from being connected to the shield plug and subsequently used.

8.4.2.4 Port 4. Port 4 is a combination port designed to accommodate the safety-class rupture disk that protects the MCO confinement boundary while the MCO is flooded and also to serve as a backup exit process port for cold vacuum drying. This port is connected to a short inlet with a 1.0-in.-diameter passageway. As a process port, the valve operator is connected to the port, and the holder of the rupture disk is the valve plug body. If this port were used as a process connection port, then any MCO pressure would have to be bled down at another process port before a process connection using the valve operator could be made at port 4. Because of the design of the rupture disk holder, the flows from the holder exit the holder laterally. In the event of a rupture disk burst with a blind cover plate installed, the cover plate is fully adequate with margin at the 450 lb/in² design pressure to contain the flow.

8.4.2.5 Port Appliances. The process valve materials are ASME SA-193 B8S or B8SA or SA-479 UNS 21800. These materials are austenitic stainless steel and are well suited for the application. The shield plug is 304/304L stainless steel and the valve C-seals are Inconel-based with noble metal seal face plating. The valve materials are compatible with the shield plug and the C-seal materials. The valve system and materials are well suited for temperatures up to 375 °C and have high structural margins based on their 150 lb/in² gauge design. The radiation levels experienced by the valve system in MCO service and storage are low and of no concern when compared to the materials' capabilities.

The expected environmental conditions for the valve systems include air, helium, argon, hydrogen, water, and water vapor. During storage and operations, pressures of up to 150 lb/in² gauge and long-term temperatures of up to 132 °C during operations and storage are acceptable. Temperatures beyond the 132 °C, but below 375 °C, during processing, off-normal events, and transients pose no difficulties for the valve system.

The valve systems are made of corrosion- and oxidation-resistant materials. In their expected environmental situations during operations and storage, all the materials are quite adequate.

The design rating of the process valves is 150 lb/in² gauge with high margins. This design is the same as the 450 lb/in² gauge design that was previously associated with the higher design pressure shield plug mechanical closure system. The designs for the plug valve and the port cover plate sealing systems have been tested hydrostatically to 600 lb/in² gauge during test programs with favorable results. As a practical matter, these plug valves are capable for 450 lb/in² gauge service even though such conditions are not anticipated.

For further discussions of the temperature and pressure capabilities of the MCO and its components, see Sections 2.2.6.2.1 and 2.2.6.2.2 of this report.

8.4.3 Port Covers

Port cover plates are available for use on the MCO. These covers serve multiple needs that may include (1) protecting appliances installed in the port pocket (Abatt 1997), (2) sealing the port, (3) providing a regulating orifice for gas leaving the MCO through the port, and (4) allowing unregulated flow from the port. A means to seal the covers to the shield plug body at the port is available. These seals would be applied in the utilization of the blind cover for sealing purposes and in the case of an orifice cover for regulating gas flows from the MCO.

Covers for the ports have two bolt patterns. The covers for the long process tube at port 3 have a five-bolt pattern. All the other port covers have a four-bolt pattern securing the cover to the MCO shield plug body. The cover attachment may be made up using a seal in between the cover and shield plug. This cover would be used when the port needs to be sealed to ensure a low leakage rate or when the appliance below needs physical protection. The blind cover is the only cover available for use on the port 3 five-bolt pattern layout.

A large hole cover is available in a four-bolt pattern. This cover would be used without a seal when the appliance below needs physical protection and the gas flows are not regulated by the 1.0-in.-diameter cover hole. The application of a large hole cover over the safety-class rupture disk to protect and allow full flow out of port 1 is planned. The large hole plate is currently planned to be used at K Basins, on the trip to the CVDF, and while the MCO is flooded with water. Preventing disk pieces and debris from getting into the orifice is considered in the design.

An optional orificed cover is available in a four-bolt pattern. This cover may be made up using a seal in between the cover and shield plug. This cover would be used when gas regulation from the port is needed. This cover comes with a 0.25-in.-diameter hole to regulate gas flow. Application of this orificed cover over the possible process pressure relief to limit the flow out of the MCO to the equivalent of a 0.25-in. orifice at port 1. An orificed plate cover is also possible on port 4, which contains the safety-class rupture disk. It is also possible for an installed appliance to serve as the restrictive orifice, thus a cover installed for orifice reasons may not be needed in all cases.

Cover plates with and without holes are available for physical protection of the rupture disk in the port. The plates provide armor in case objects are dropped onto the MCO. The large hole cover has a 1-in.-diameter hole in it, and when installed in the port, the system allows for full disk operation and matches the flow capacity of a burst rupture disk. An installed blind cover plate with a "C" seal and bolts effectively seals the port.

8.4.4 Multi-Canister Overpack Cover Cap

The MCO will be equipped with a cover cap that goes over the large end of the MCO. This cap is welded to the shell collar and effectively covers the four ports in the shield plug and limits the use of these ports. Once the cover cap is installed, configuration changes to the covered ports are not possible. The cover cap design contains a penetration that can be used to relieve pressure, replace the gas accumulated under the cover cap, and accommodate leakage rate testing of the cover cap weld. Other uses for this penetration are possible. The MCO design pressure of 150 lb/in² gauge increases to 450 lb/in² gauge when the cover cap is welded onto the MCO.

8.4.5 Maintenance

Before the cover cap is welded in place, the maintenance needs of the MCO are centered around the four ports and the mechanical features used to support the functions of the ports. The ability to use the port covers and mechanical operators on all the ports at the shield plug is necessary. Preservation of the sealing face and the use of the four or five anchoring bolt holes is important to this task. Preservation of the threads securing the appliance into the port also is necessary and includes the repair of those threads. This may include installing thread inserts and doing machining to remove seized appliances. Preservation of the sealing face for the appliance seal is also needed.

On port 1, it is possible to remove and replace various appliances that may serve the process pressure relief function. Ports 2, 3, and 4 involve the process valves and the safety-class rupture disk. During operations at ports 2 and 3, the valves are not normally removed from the shield plug. However, on ports 2, 3, and 4, the ability to remove and replace the valve plug is necessary. Any needed repairs to removed plugs would likely be done offline. Change out of a seal on the process valves or cover ports may be accomplished at the shield plug during the

remove-and-replace operation. After welding of the cover cap and placement into interim storage, no maintenance of the MCO is required.

8.4.6 Multi-Canister Overpack Process Step Descriptions

Table 8-1 shows the lineup of the MCO ports. It describes which ports are open or closed and variances during the process path. This table is intended to show only the opening and closing cycles for the process valves and the major changes occurring at the port.

Table 8-1. Multi-Canister Overpack Port Lineup.

Port number	Sequence steps, appliance or condition ^a				
	1 Transport to CVDF	2 Process operations at CVDF	3 Transport to CSB	4 Receipt verification at CSB	5 ^b MCO weld
1	C	C	C	C	C
2	O	O	C	O	C
3	C	O	C	C	C
4 R ^c	C	C	C	C	C

^aSequence steps are tied to steps in Table 8-2.

^bThis step concludes with installation of the cover cap on the multi-canister overpack.

^cAt port 4, a safety-class rupture disk is installed in the plug valve, and the disk is always active to the port cavity.

C = closed valve or port.

CSB = Canister Storage Building.

CVDF = Cold Vacuum Drying Facility.

MCO = multi-canister overpack.

O = open valve or port.

R = rupture disk installed.

The configuration of the MCO ports at different steps in the process sequence leading to the MCO being put into sealed interim storage is summarized in Table 8-2. The numbering of the sequence steps in Table 8-1 corresponds to the numbering of the steps in Table 8-2. To determine the configuration change and needed effort along the sequence, compare the MCO port states in each step and determine changes necessary to make the configuration agree with the step.

Table 8-2. Life Cycle Steps of the Multi-Canister Overpack. (3 sheets)

Location of MCO	Port number	Configuration	Active list
Step 1: Transit to CVDF			
<p>The MCO is in the cask at a K Basin load-out pit. The shield plug and locking and lifting ring are installed. The set screws have been tightened and the cask lid is installed.</p> <p>The MCO is vented to the interior of the cask. The cask lid is sealed. The fuel inside the MCO is flooded with water, and there is inert gas above the fuel beneath the shield plug. The cask is on the conveyance.</p>	1	The port is plugged and the cover plate is installed.	This port is considered inactive.
	2	The process valve is open and a large-hole cover plate is installed.	This port is active and allows the MCO to vent through the internal HEPA filter into the cask cavity.
	3	The process valve is closed and a blind cover plate is installed without a seal.	This port is inactive.
	4	The safety-class rupture disk is installed in the port and the port is covered with a large-hole cover.	This port is active to relieve MCO internal pressure in the event of a safety-class rupture disk actuation.
Step 2: Process operations inside CVDF			
<p>The MCO is brought into the CVDF in the cask and on the conveyance; the operating equipment is attached to the process ports and the relief port is also connected to a venting system.</p>	1	The port is plugged and the cover plate is installed. This port is a preferred alternative port if the HEPA filter is not damaged.	This port is considered inactive.
	2	This port is connected to the process system and is first used to introduce inert gas into the MCO during the bulk water removal cycle, then used to vacuum dry the fuel and evacuate other gases from the interior of the MCO.	The port is active, covers are removed, and the port valve is open.
	3	This port is connected to the operating system and is first used to remove the bulk water from the MCO. Then the port also may be used for the introduction of gases into the MCO.	The port is active, covers are removed, and the port valve is open.
	4	The safety-class rupture disk is installed in the port. If the MCO internal filter becomes inoperable, this port also serves as an alternate backup process exit port for port 2. Use of this port allows unfiltered flow into the CVDF. Cover plates are removed at the facility.	Rupture disk is active.

Table 8-2. Life Cycle Steps of the Multi-Canister Overpack. (3 sheets)

Location of MCO	Port number	Configuration	Active list
Step 3: Transport to CSB			
The MCO is in a sealed cask and the MCO is on the conveyance.	1	The port is plugged and the cover plate is installed.	The port is considered inactive.
	2	This port is configured with a closed valve and a sealed blind cover plate.	Port is inactive.
	3	This port is configured with a closed valve and a sealed blind cover plate.	Port is inactive.
	4	The rupture disk is installed and the port is sealed with a blind cover plate.	The rupture disk is considered inactive.
Step 4: Process verification operations on receipt of MCO at the CSB			
The MCO is at the weld station of the CSB. The as-received configuration is described. Any operations performed at this step are roughly described under port 2 configuration.	1	The port is plugged with a sealed blind cover.	The port is considered inactive.
	2	This port is configured with a closed valve and a blind cover. The blind cover and seal are removed. A sampling operator is attached to the MCO. The MCO is gas sampled. The operator is removed, and the blind cover plate and seal are installed.	Port is active during sampling.
	3	This port is configured with a closed valve and a blind cover with seal.	Port is inactive.
	4	The rupture disk is installed and the port is sealed with a blind cover plate and seal.	The rupture disk is considered to be inactive.

Table 8-2. Life Cycle Steps of the Multi-Canister Overpack. (3 sheets)

Location of MCO	Port number	Configuration	Active list
Step 5: CSB MHM handling of MCO to welding			
<p>The MCO is brought to the weld station from a CSB tube or from the receiving pit. The MCO is prepared for welding at this station, and then the cover cap is welded to the MCO. Actions taken here may include the following.</p> <ul style="list-style-type: none"> • Sample interior gas of MCO. • Leakage rate test the MCO mechanical main seal. • Configure the ports for cover cap welding. • Locate and weld the cover cap. • Perform weld examination as required. • Seal up cover cap penetration, perform leakage rate tests. 	1	This port has already been configured with a threaded plug that has a mechanical seal ring. This port will have a sealed cover plate over it when the cover cap is installed.	The port is inactive.
	2	After welding, this port is configured with a closed valve. This port may not have a sealed cover plate over it when the cover cap is installed.	Port is inactive.
	3	After welding, this port is configured with a closed valve. The port is covered with a blind cover plate and mechanical seal.	Port is inactive.
	4	The rupture disk is installed and the port is sealed with a blind cover plate and seal	The rupture disk is considered inactive.

CSB = Canister Storage Building.
 CVDF = Cold Vacuum Drying Facility.
 HEPA = high-efficiency particulate air (filter).
 MCO = multi-canister overpack.
 MHM = multi-canister overpack handling machine.

9.0 ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM

9.1 ACCEPTANCE CRITERIA

No maintenance or in-service inspection of the multi-canister overpacks (MCOs) is required while they are in normal service from the K Basins through interim storage at the Canister Storage Building (CSB). Process inspection and/or examination steps are planned onsite to ensure the MCO is properly configured and processed for interim storage. The following four steps are performed at the Cold Vacuum Drying Facility (CVDF) and CSB.

1. After process drying at the CVDF, a combination of tests and procedures is performed to ensure the MCO residual water inventory is within allowable limits. The test procedure basis is described in HNF-1851, *Cold Vacuum Drying Residual Free Water Test Description* (Pajunen 1998a). The procedure is initiated by holding the MCO slightly pressurized with helium for at least 4 hours to equalize fuel and scrap temperatures. A pressure rebound test is then performed to demonstrate adequate surface water removal. Once the rebound test is satisfied, the MCO is held at a predefined set of conditions (<12 torr pressure, >40 °C temperature) for between 8 hours and 28 hours, depending on the number of scrap baskets. Successful completion of these procedure steps allows one to conclude the MCO free water content is less than 200 g. A second rebound test is performed at the conclusion of the drying procedure to demonstrate no inadvertent water additions have occurred during the drying procedure. The tests and procedures ensure that the MCO has been properly dried at the CVDF.
2. Upon completion of cold vacuum drying processing, the MCO's mechanical seals are tested to ensure that they meet a 1×10^{-5} standard $\text{cm}^3/\text{s-atm}$ total integrated leakage rate test acceptance criterion, as specified by HNF-S-0426, *Performance Specification for the Spent Nuclear Fuel Multi-Canister Overpack* (Goldmann 2000b) and defined by ANSI N14.5-1987, *For Radioactive Materials — Leakage Tests on Packages for Shipment* (ANSI 1987). That acceptance criterion, which is based on the functions and requirements and operational leak rate calculations provided in HNF-2155, *Multi-Canister Overpack Combustible Gas Management Leak Test Acceptance Criteria* (Sherrell 1999) will ensure that a positive helium gauge pressure will remain (at temperatures down to -31 °C) within any mechanically sealed MCO for well over 40 years without replenishment, and that, based on either barometrically induced air exchange alone or diffusion alone (both mechanisms will actually be active), the hydrogen concentration within a vented CSB storage tube containing two mechanically sealed MCOs will never exceed 1 vol%. Finally, based on a very conservative (i.e., large) allowance for the hydrogen inventory of an MCO during shipment from the CVDF to the CSB, HNF-SD-TP-SARP-017, *Safety Analysis Report for Packaging, Onsite, Multi-Canister Overpack Cask* (Smith 2000), calculates that the 1×10^{-5} standard

cm³/s-atm integrated leak test acceptance criterion will provide a 234-hour dry transfer window for that shipment.

3. After welding the cover cap onto the MCO's collar at the CSB, the welded joint and the entire cover cap are leakage rate tested to ensure that the MCO meets the 1×10^{-7} standard cm³/s-atm total integrated leakage rate test acceptance criterion specified by HNF-S-0426 (Goldmann 2000b). In addition to ensuring that the MCO welds are adequate to meet all current and foreseeable needs of the SNF Project for combustible gas management, that acceptance criterion also ensures that the MCO is leaktight, as defined in ANSI N14.5-1987 (ANSI 1987).
4. The cover cap's attachment weld is also dye-penetrant inspected, as a minimum, after the root and final weld passes. This examination ensures the examined weld passes are of proper surface quality and is done in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (ASME 1998) under Code Case N-595.

The MCOs are ASME stamped vessels, and as such, a great deal of inspection, examination, and testing is performed on the components prior to their ASME Code stamping as part of the MCO pressure boundary. Routine material and fabrication evaluations performed on the MCO and then documented include certified material test reports, radiography, ultrasonic examination, helium leakage rate tests, dye-penetrant inspections, and hydrostatic pressure testing. These evaluations are needed to ASME stamp the MCO and are performed in accordance with the ASME Boiler and Pressure Vessel Code (ASME 1998) requirements. Test pressures, acceptable leakage rates, equipment arrangements, standards, examiner requirements, and other requirements and acceptance criteria are provided for in the ASME Boiler and Pressure Vessel Code (ASME 1998). Later in this chapter, in Section 9.1.1, discussion is provided describing some of the ASME Boiler and Pressure Vessel Code activities performed during the fabrication of the MCOs.

Although not part of the safety basis, the MCOs are monitored at the CSB in the areas of gas sampling, temperature, and pressure as described below.

- Several MCOs will have their interior gas periodically sampled, and the gas sample will be sent off for analysis. The MCOs so sampled will not have their cover caps welded onto the collar. When the gas sampling program is complete, the cover caps may then be welded onto the sampled MCOs and these MCOs sealed, if desired.
- Several MCOs will have a thermometer installed at the top of the shield plug so the shield plug temperature may be read while the MCO resides exposed in a CSB storage tube. This task is still being planned and details are being developed. The MCOs that are being temperature monitored will not have their cover caps welded onto the collar. When the temperature monitoring program is complete the cover caps may then be welded onto the monitored MCOs and these MCOs sealed, if desired.

- The pressure monitoring plan provides for all MCOs being equipped with pressure monitoring units while the MCOs are in storage at the CSB. These units will function with and without cover caps installed on the MCO. The details of this pressure monitoring program are currently incomplete. The plan is to periodically pressure-monitor an exposed MCO while the MCO is in a CSB storage tube. It is expected that the plan will not affect or impair the MCOs' performance during storage.

9.1.1 Visual Inspections and Nondestructive Examination

The fabrication specification for the MCO (Goldmann 1998) describes the examinations to be performed by the fabricator of the MCO. The fabrication specification requires the fabricator to submit a written manufacturing plan and schedule, and to submit written and approved examination procedures and reports for buyer approval. The drawings appended to the fabrication specification (Goldmann 1998) identify the welds to be examined and the type of examinations to be performed, and the specification text identifies the acceptance criteria. These requirements are presented below.

Personnel performing nondestructive examinations shall be qualified in accordance with NB-5520 or NG-5520, as applicable, ASNT SNT-TC-1A, and ANSI N45.2.6. Only individuals qualified for nondestructive testing levels I, II, or III may perform nondestructive testing. Personnel qualified at level I shall not interpret the results of an examination or make a determination of the acceptability of an examined part. Examinations shall be performed in accordance with Section V, Articles 2, 6, 7, and 9 for radiographic, liquid penetrant, magnetic particle, and visual methods, respectively.

All welds made by the fabricator shall be visually examined in accordance with the requirements of NB-4424 and NG-4424 as required. These requirements specify that welds that fail to meet the requirements will be repaired as necessary and re-examined.

The weld joining the cylindrical shell of the MCO to the bottom baseplate is a full-penetration circumferential weld that is to be examined by radiography, with acceptance determined according to the criteria of NB-5320 of the *Boiler and Pressure Vessel Code* (ASME 1998), and by liquid penetrant examination, with acceptance determined according to the criteria of NB-5350 (ASME 1998). Defects in weld metal detected by these examinations will be repaired and the repair examined in accordance with NB-4450.

Welds in the fuel baskets and associated support structures shall conform to the requirements of NG-4000. Some welds require liquid penetrant examination, with acceptance determined according to NG-5350. Defects in weld metal detected in these examinations shall be repaired and the repair examined in accordance with NG-4450.

The weld joining the cylindrical shell to the collar is a full-penetration circumferential weld. Examination shall be by radiography, with acceptance determined according to NB-5320.

(ASME 1998), and by liquid penetrant examination, with acceptance determined according to the criteria of NB-5350 (ASME 1998). Defects in weld metal detected in these examinations shall be repaired and the repair examined in accordance with NB-4450.

The MCO and shell assembly welds will be radiographically examined, and the mechanical closure assembly, except for the actual production Helicoflex seal (see Section 9.1.1.2), will be hydrostatically tested at the fabricator's plant. These main structural parts of the MCO assembly outside of the shield plug penetrations, ports, connectors, and port covers will have no known exceptions to the Boiler and Pressure Vessel Code (ASME 1998), Subsection NB.

With application of ASME code case N-595-1, the cover cap portion of the MCO also will comply with Subsection NB with no known exceptions. For closure welds on spent fuel containers, this code case permits multipass dye penetrant inspection in lieu of volumetric examination and also exempts the final closure weld from pressure testing requirements.

9.1.1.1 Shop Pressure Tests. The mechanical closure and shell assembly will be hydrostatically tested, then helium leakage rate tested as an assembled unit at the fabricator's plant. While at the basin load-out pit, the assembled MCO, with the shield plug and locking and lifting ring installed, will have the 18 set screws torqued down. As this will be a qualified assembly, the goal is to make only a routine examination or leakage rate determination at the K Basins if required. Leakage rate testing of the MCO's main seal (Helicoflex) joint will occur at the CVDF.

9.1.1.2 Shop Leakage Rate Tests. After the welds joining the shell to the MCO bottom and collar have been completed, the entire surface of the shell, welds, collar, and bottom end cap shall be helium leak tested. The leak test shall be performed in accordance with ANSI N14.5-1987 fabrication test requirements (ANSI 1987) and the ASME Code, Section V, Article 10, Appendix V, "Hood Technique" (ASME 1998). The leakage test shall be performed using a pressure differential of 1 atm. The maximum acceptable leakage is 1.0×10^{-7} standard $\text{cm}^3/\text{s}\cdot\text{atm}$. The leak test system must be calibrated with a National Institute of Standards and Technology (NIST) traceable calibrated standard in the range of $\times 10^{-7}$ standard $\text{cm}^3/\text{s}\cdot\text{atm}$. The leak standard shall be positioned in the system such that it represents the farthest point (at a minimum) the helium must travel to reach the helium mass spectrometer leak detector. The helium standard must be continuously evacuated by an auxiliary pump just before being released into the system. The helium response time and system sensitivity shall be documented. The test procedure shall include the method used to ensure that the helium concentration inside the hood or enclosure is approximately 100% across the entire surface of the MCO shell, weld, and bottom end cap.

The helium leak test of the first MCO fabricated shall demonstrate that the MCO main metallic seal and the test port cover metallic seals are capable of sealing to a maximum leakage rate of 1×10^{-5} standard $\text{cm}^3/\text{s}\cdot\text{atm}$ at a pressure differential of 1 atm. The total seal leakage (main seal plus test port covers) of the MCO shall be less than 1×10^{-5} standard $\text{cm}^3/\text{s}\cdot\text{atm}$ at a differential pressure of 1 atm. The test shall be conducted so that any leakage is from inside of the MCO to the outside. After the first MCO, it is permissible to use elastomeric seals in place of the metallic seals. When using elastomeric seals, the use of vacuum greases shall be minimized. Vacuum greases shall not be silicone-based materials. The leak test procedure shall describe in

detail the seal testing sequence and test parameters. The leak test procedure shall be performed in accordance with ANSI N14.5-1987 fabrication test requirements (ANSI 1987) and the ASME Code, Section V, Article 10, Appendix V, "Hood Technique" (ASME 1998). The leak test system must be calibrated with an NIST-traceable calibrated standard in the range of $\times 10^{-4}$ or $\times 10^{-5}$ standard $\text{cm}^3/\text{s}\cdot\text{atm}$. The test procedure shall include the method used to ensure that the helium concentration on the inside of the MCO is approximately 100%.

After the MCO cover cap has been machined, but before the cap penetration has been bored, the cover cap shall be helium leak tested. The leak test shall be performed in accordance with ANSI N14.5-1987 fabrication test requirements (ANSI 1987) and the ASME Code, Section V, Article 10, Appendix V, "Hood Technique" (ASME 1998). The leakage test shall be performed using a pressure differential of 1 atm. The maximum acceptable leakage is 1×10^{-7} standard $\text{cm}^3/\text{s}\cdot\text{atm}$. The leak test system must be calibrated with an NIST-traceable calibrated standard in the range of $\times 10^{-7}$ standard $\text{cm}^3/\text{s}\cdot\text{atm}$. The helium standard must be continuously evacuated by an auxiliary pump just before being released into the system. The helium response time and system sensitivity shall be documented. The test procedure shall include the method used to ensure that the helium concentration inside of the hood or enclosure is approximately 100% across the entire inner surface of the machined MCO cover cap.

Requirements regarding the use of cleaners and solvents during fabrication are presented in HNF-0453, *Specification for Multi-Canister Overpack Fabrication* (Goldmann 1998). Section 13.6.3 describes the requirement for the supplier to report defects and noncompliances.

9.1.2 Components

The following component and material groupings are discussed in this section:

- Appliances and associated materials installed in the ports
- The port covering system and associated materials used in the port cover positions
- MCO internal filters
- The MCO main seals.

The port appliances include the plug valves and rupture disk assembly. Also included are the metallic seals used to seal the appliances to the shield plug body. The individual manufactured pieces are dimensionally inspected by the manufacturer and, as required, tested by the manufacturer in accordance with the ASME Code (ASME 1998). The completed assembly is installed in the shield plug and leakage rate tested by the MCO fabricator. The seals used for the plug valves are reusable because of the crush-limiting design of the assembly. Seals for the plug at port 1 are not normally reusable and would require replacement should the need arise to change out the appliance at MCO port 1 and effect a seal.

The port cover grouping includes the covers, cover bolts, and cover seals. All the covers are inspected dimensionally by the manufacturer. The blind cover is also leakage rate tested at the factory with a new seal installed. The port cover seals are not normally reusable, so when a seal is required in the cover assembly, a new seal is installed.

The MCO is weld-sealed in the Canister Storage Building. Even though no contamination is expected, the MCO has internal high-efficiency particulate air (HEPA) filters to reduce the potential for contamination to be released. The MCO internal filters are used for process purposes and are HEPA-filter rated. The HEPA filter bank assembly is first assembled with the shield plug, and then the shield plug guard plate is installed on the shield plug assembly. After the shield plug assembly is complete, the efficiency of the filter bank assembly is tested by the MCO fabricator to verify HEPA filter performance.

The MCO main seal interfaces with the MCO shell collar and the shield plug. This sealing interface is dimensionally inspected and tested by the MCO fabricator. Any seals used at the factory to test the sealing interface are not reused in the MCO final assembly in the K Basin load-out pits. With the metal main seal not being reusable, a new MCO main seal is used in the MCO assembly for MCO closure operations at K Basins. The new main seals come directly from the manufacturer in sealed packages for use at the K Basins.

The cover cap attachment weld is leakage rate tested to pass a leakage rate of 1×10^{-7} standard $\text{cm}^3/\text{s}\cdot\text{atm}$ at the Canister Storage Building. This leakage rate test is done immediately after welding and before the dye penetrant test is performed on the last pass of the weld. Depending on how the cover cap weld is inspected, the penetration threaded mechanical seal may be tested separately. If it is tested separately it will be inspected to be less than a leakage rate of 1×10^{-7} standard $\text{cm}^3/\text{s}\cdot\text{atm}$. Testing is conducted on a pass or fail basis, and no sensitivity band is used.

9.1.3 Shielding Integrity

The shield plug is the only MCO component that provides a specific shielding function. The results for the shielding evaluation, presented in Chapter 5.0, are acceptable.

9.1.4 Thermal Acceptance

Thermal acceptance of the MCO closure system is based on prototype testing and design verification. Extensive prototype testing was performed on the closure system, and the system was found to be acceptable under the specified thermal cycling conditions (Crow 1996). Also, thermal evaluations conducted by the design agent, Parsons Infrastructure and Technology, show that the design is acceptable (Goldmann 2000a). As such, additional thermal testing of the closure system for the production MCOs is not needed. Acceptance is based upon the prototypic test results and analyses.

9.2 MAINTENANCE PROGRAM

No planned maintenance program or preventative maintenance activities are expected for the MCO or the process plug valves and port covers. The rupture disks themselves do not require maintenance; however, replacement of the disk assembly would be performed if the disk actuates in service.

Corrective maintenance will be based on needs occurring in service because of damage or unforeseen causes. Potential hazards include flammable gas buildup and release through any port and the potential for unfiltered exposure and contamination from ports 3 and 4. The MCO may require repair of components associated with the four ports located in the top of the shield plug. The repair may include removal and/or replacement of appliances, replacement of the metal seals on the appliances, repair of appliance or cover bolt threads, or repair of sealing faces for ports or appliance seals. Equipment may be necessary to perform these operations and repairs.

Port 1 is currently blank and plugged, and removal and/or replacement of appliances in this port is possible. Normally, the process valve plugs associated with ports 2, 3, and 4 are not removed from the shield plug, so equipment will have to be developed to accomplish removal and/or replacement at these ports. Additional details are provided in Section 8.4.5.

The port appliances such as rupture disk assembly, process plug valves, and plugs are replaced from the outside, at the top, of the MCO. Also the cover plate seals are replaced from the top of the MCO by removing the cover plate, replacing the seal in the cover plate cavity, and then rebolting the cover to the shield plug port opening. While such "replacements" are possible, no maintenance is planned or expected. No components inside the MCO require maintenance or replacement under normal operating conditions. The MCO components are all produced and inspected in accordance with the fabrication specification and drawings prior to their shipment to the Hanford Site. The fabricator tests include dimensional, welds, and leakage rate testing. During process onsite the MCOs are designed so leakage rate testing may be performed as appropriate for the main seal, port cover plates, cover cap attachment weld, and the cover cap penetration. Also the cover cap attachment weld has been designed to accommodate ultrasonic testing and all the welds associated with the cover cap installation are capable of being dye penetrant tested. Testing of safety class components inside the MCO is not required. All safety-class components have been designed to fully consider all required maintenance for the port to fulfill its intended functions. No surveillance testing requirements are relied upon to confirm that the engineered safeguard features will operate during the accident conditions that have been analyzed.

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10.0 RADIATION PROTECTION

10.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE

10.1.1 Policy Considerations

The multi-canister overpack (MCO) design, in combination with other equipment and facilities, incorporates features to provide radiological protection and control and to support the basic philosophy of reducing radiation exposure levels to values as low as reasonably achievable (ALARA). The MCO design takes into consideration planned inspections, handling, and possible repair. Design, handling, and operations will be consistent with Project Hanford radiation protection policies and procedures and Title 10, *Code of Federal Regulations*, Part 835, "Occupational Radiation Protection" (10 CFR 835) requirements (see Section 2.3.5).

10.1.2 Design Considerations

Careful consideration has been directed towards designing the MCO to ensure that appropriate radiological control features are incorporated and that occupational radiation exposure from MCO operations will be ALARA (see Chapter 5.0). The MCO needs to effectively reduce radiation exposure levels and also to perform its intended purpose efficiently. The MCO design uses connections that provide for remote manipulator operation or connection via long-handled tools to minimize, if required, the time during which a worker's hands are in proximity to the MCO shield plug. Sealing the MCO mechanically results in an estimated personnel dose of 195.6 person·mrem per MCO at the K East Basin and 117.6 person·mrem per MCO at the K West Basin (Goldmann 1996b). These doses presume a 1 in 100 mechanical repair ratio (i.e., one MCO in every 100 will require one additional handling for repair and result in additional dose). For a total campaign of 400 MCOs, split equally between the two basins, a total personnel dose of 46 person·rem is spread out over the 2-year period.

Radiation shielding features have been incorporated into the MCO design. The MCO shield plug protects workers from photons and neutrons emanating from inside the MCO. This shielding will achieve an average dose rate across the top of the shield plug of 2 to 4 mrem/h on contact (within 2 in.) for the average MCO fuel material to be handled. This value includes radiation streaming between the MCO shield plug and the MCO shell and around penetrations. For the worst-case spent nuclear fuel to be handled in the MCO, the average dose rate across the top of the shield plug on contact (within 2 in.) is less than 8 mrem/h (see Table 5-7 and associated note). By reducing the average intermittent radiation dose rate to 2 to 4 mrem/h for very brief "hands-on" operations, the extremity dose is minimized while maintaining a practical and efficient design requiring minimal maintenance. The MCO design also has been influenced strongly by the desire for low maintenance, which translates into less overall radiation exposure for plant operations.

The MCO design allows for the spent nuclear fuel to be returned to the K Basins should the MCO be leaking, defective, or damaged. The MCO shell design includes a machined plate with a small liquid collection sump for water removal, and features and devices to facilitate loading the spent fuel baskets.

10.1.3 Operational Considerations

The MCO design includes features on the shell and shield plug for safe lifting and handling of MCO components and loaded MCOs. Hands-off handling of MCOs with lifting devices is a key feature in reducing overall radiation exposure.

Facility-specific operating procedures and training will have a primary focus on safety and radiation control (and ALARA). Controlling radiation exposure depends on the careful coordination of handling operations and the minimizing of residence time in radiation fields. Management emphasis on radiation protection and ALARA in team-building situations such as procedures development and training will help reinforce the appropriate safety culture and ALARA awareness for MCO operations.

10.2 RADIATION PROTECTION DESIGN FEATURES

Design of the MCO also has been driven by the need for a high-integrity, high-reliability container. The MCO main seal and cover seals are leakage-rate tested before the MCO leaves the Cold Vacuum Drying Facility. The leakage rate acceptance criterion for these tests is 1×10^{-5} atm-standard m^3/sec . At the Canister Storage Building the cover cap weld is leakage-rate tested and the criterion for acceptance is 1×10^{-7} atm-standard m^3/sec .

The exposed surfaces of the MCO will be smooth to the greatest extent practicable to minimize contamination. Corners and other features that could collect contamination have been minimized. Future access for gas sampling is facilitated by an internal bank of four high-efficiency particulate air (HEPA) filters.

10.3 ESTIMATED ONSITE COLLECTIVE DOSE ASSESSMENT

Planned MCO operations on a facility-specific basis are reviewed to determine potential onsite dose estimates associated with major functions such as storage, handling, processing, or inspections. Facility-specific operations will factor estimated dose rates into overall facility operations and radiation protection management. Detailed dose assessments will be performed outside of this topical report to facilitate appropriate operational planning activities.

10.4 AS LOW AS REASONABLY ACHIEVABLE CONSIDERATIONS

ALARA principles and good practice were used in the course of designing the MCO. The design of the MCO collar includes a tapered section on the outside near the bottom of the shield plug assembly to prevent and minimize line-of-sight scattered, direct, or fluoresced radiation from coming out of the annulus created outside of the MCO. The cask design uses this step taper to create a labyrinth to minimize the dose rates emitted from the created annulus to achieve ALARA.

The interior of the MCO includes a bank of HEPA filters for exiting process gases to minimize the radiological contamination risk to personnel around the MCO during the making and breaking of process connections on the top of the shield plug and during process operations in the facilities and using associate systems. This process bank of filters will retain as much of the process gas-swept contamination inside the MCO as possible and keep it behind the shield plug main body to reduce operator dose potential. Ports that are not HEPA filtered are preceded by screens that retain particles greater than 2 mm in diameter inside the MCOs. These filtering and screening features further demonstrate support of ALARA principles.

The design of the MCO shield plug reflects numerous features selected to achieve ALARA. The thickness of the stainless steel plug was maximized as much as possible to minimize the dose at the top of the shield plug. Examples of compliance with Regulatory Guide 8.8, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable* (NRC 1978), include passageways through the shield plug with 90° angles to prevent streaming (NRC 1978, Section 2.b.4), and shield plug appurtenances were designed to be actuated via both remote-operated and long-handled tooling to reduce dose by means of distance (NRC 1978, Section 2.b.3). The selection of the mechanical closure was based, in part, on study results that included detailed dose per worker hour estimates for the specific operation steps involved in closing the MCO and cask.

The MCO project also has produced other shielding studies, dose maps, and reports to support the systems, facilities, and operations that will use and interface with the MCO and hardware. This information is published in design reports, engineering studies, and shielding studies. The pertinent MCO specifications, reports and document numbers are listed below. Much of the initial and preliminary work conducted to determine direction is not included in these references:

- HNF-S-0426, *Performance Specification for Spent Nuclear Fuel Multi-Canister Overpack* (Goldmann 2000b)
- HNF-SD-SNF-CN-026, *MCO Shield Plug Dose Rate Analysis* (Hillesland 1997)
- HNF-SD-SNF-DR-003, *Multi-Canister Overpack Design Report* (Goldmann 2000a)
- WHC-SD-SNF-ES-021, *Multi-Canister Overpack Closure Comparison Study* (Goldmann 1996b).

These paragraphs document the MCOs' compliance with the applicable requirements of 10 CFR 835. The facility, administrative procedure, and administrative control areas' compliance with applicable portions of 10 CFR 835 has to be, by its nature, performed by others outside of the MCO subproject.

11.0 MULTI-CANISTER OVERPACK ACCIDENTS

11.1 INTRODUCTION

The multi-canister overpack (MCO) is the primary container for spent nuclear fuel (SNF) from the time the fuel is packaged in the K Basins through its placement into interim storage at the Canister Storage Building (CSB). An MCO functions as a container during transportation of the SNF, as a process vessel during conditioning of the SNF at the Cold Vacuum Drying Facility (CVDF), and as a storage vessel in the CSB. An MCO is the primary confinement boundary for SNF throughout the process cycle. The purpose of this chapter is to present the technical basis for determining the classifications of MCO structures, systems, and components (SSCs), which are summarized in Table 11-1.

Facility hazard analyses for the K Basins (Strickland 1994), CVDF (Powers 1998b), and CSB (Powers 1998a) identified credible event sequences that could challenge the structural integrity of an MCO. The forces that challenge an MCO during an accident may be internal or external to the MCO. Internal forces are similar to external forces to the extent that each introduces the possibility of MCO failure and subsequent release of SNF. Internal forces differ from external forces, however, in that internal forces are inherent to the MCO and SNF, and external forces are imposed on the MCO and SNF. External forces are not inherent to an MCO and are not covered in this chapter. External forces that act on the MCO will be fully addressed in the facility-specific safety analysis reports.

The accidents selected for analysis are derived from the hazard analyses performed for the K Basins (Chaffee et al. 1999), CVDF (Powers 1998b), and CSB (Powers 1998a). The hazard analysis process is a systematic examination of planned activities involving an MCO in all the areas of each facility. The hazard analyses identified scenarios that could result in releases of hazardous materials from an MCO. Bounding, representative, and unique upset conditions were selected from groups of hazardous conditions with similar release characteristics and event initiators. Those selected upset conditions that are inherent to an MCO and could be realized at various times during the life of an MCO are the subject of this chapter.

From the hazard analyses, the accidents selected as inherent to an MCO are (1) a criticality event, (2) an MCO breach caused by a thermal runaway reaction, (3) hydrogen deflagration or detonation in an MCO, and (4) overpressurization events and continuous releases (leaks).

Fuel ignition has been shown to be incredible (Plys et al. 1998a). Therefore, fuel ignition is not addressed in this chapter. Furthermore, there is no postulated hydrogen detonation following a thermal runaway, so no analysis was performed.

Table 11-1. Safety Functions and Failure Consequences of Multi-Canister Overpack Structures, Systems, and Components. (2 sheets)

System or component	Function	Safety designation	Failure consequences	Accident
Shell assembly, including collar, shell, and bottom plate	Contain and protect SNF	SC	Release of radioactive contents that could exceed offsite exposure limits; loss of contingency protection against nuclear criticality accident	Overpressurization Criticality
Shield plug	Contain SNF, protect personnel	SC	Release of radioactive contents that could exceed offsite exposure limits	Overpressurization Criticality
Rupture disk	Protect MCO pressure boundary	SC	Damage or expansion of MCO boundary	Overpressurization Criticality
Locking and lifting ring and bolts	Maintain pressure on main seal, allow for lift of loaded MCO	SC	Release of radioactive contents that could exceed offsite exposure limits	Overpressurization
Cover cap	Seal MCO	SC	Release of radioactive contents that could exceed offsite exposure limits	Overpressurization
Mark IA baskets, structural stainless steel portion	Maintain Mark IA SNF elements and scrap in a safe configuration	SC	Loss of double-contingency protection against nuclear criticality accident	Criticality
Mark IV baskets	Hold Mark IV SNF elements and scrap	GS	No release consequences	NA
Mark IA and IV scrap basket copper shroud, copper subassembly	Hold scrap and dissipate internally generated heat within the basket	SS	No release consequences; copper is not a structural component and is used for thermal conductivity in the assembled MCO	Thermal runaway reaction
Plug valves	Accommodate gas flows through process ports in support of MCO processing	SS	Inability to process the MCO; release of radioactive materials into the environment that exceeds exposure limits	Gaseous release
Internal process filter	Maintain most radioactive solid materials within the MCO	SS	Release of radioactive materials from the MCO; pressure buildup within the MCO; loss of defense-in-depth protection for release of radioactive materials	Not credited in accident analyses
Long process tube	Allow bulk water removal, introduction of gases during processing, and reflooding, if necessary	SS	Inability to remove water from MCO; inability to introduce gases to process MCO; inability to perform processing that puts the MCO into a safe configuration	Hydrogen explosion

Table 11-1. Safety Functions and Failure Consequences of Multi-Canister Overpack Structures, Systems, and Components. (2 sheets)

System or component	Function	Safety designation	Failure consequences	Accident
Seal	Seal MCO shield plug to shell	SS	Release of radioactive materials from the MCO; pressure buildup within the MCO; loss of defense-in-depth protection for release of radioactive materials	Gaseous release
0.25-in. orifice cover plates	Regulate gas flow from the MCO	SS	Valve with rupture disk becomes nonoperational, causing possible release	Gaseous release
Cover or orifice plate bolts	Maintain seal pressure	SC	Loss of double contingency protection against leakage	Gaseous release
Cover plates	Provide leaktight seal, protect appliances	SC	Loss of double contingency protection against leakage	Gaseous release
Guard plate on shield plug	Protect internal MCO process filter, and rupture disk inlet	SC	Potential damage to filter and blockage to inlet; failure to center basket center posts	Hydrogen explosion Criticality
Port valve/plug and cover plate seals	To seal appliances to shield plug	GS	Loss of pressure; potential release	NA

GS = general service.

MCO = multi-canister overpack.

NA = not applicable.

SC = safety class.

SNF = spent nuclear fuel.

SS = safety significant.

11.2 CRITICALITY EVENT

The MCO performs the function of maintaining the SNF in a subcritical geometry. Failure of an MCO or component and subsequent movement of SNF to an unfavorable or unanalyzed geometry is a violation of criticality safety. The requirement for criticality safety is a safety-class function, as stated in Section 1300-3.2 of DOE Order 6430.1A, *General Design Criteria*, and HNF-PRO-704, *Hazard and Accident Analysis Process*. Therefore, an MCO and the associated components that have a criticality safety function — the shell assembly, including shield plug assembly, the stainless steel structural components of the Mark IA baskets, and the rupture disk — are identified as safety-class components before any estimation of accident consequences.

11.3 MULTI-CANISTER OVERPACK BREACH CAUSED BY THERMAL RUNAWAY REACTION

A thermal runaway reaction is the rapid oxidation of uranium metal occurring in an MCO if the temperature of the SNF is not controlled. Although the oxidation reaction occurs continuously when water is present in an MCO, the rate of oxidation is controlled because the MCO is maintained at a low temperature. If the fuel temperature control were lost and sufficient water were present, the oxidation reaction rate in the MCO would increase. Because the reaction is exothermic, it liberates heat, increasing the temperature in the MCO and further accelerating the reaction. As the temperature and pressure inside the MCO increase, the partial pressure of water vapor also increases, resulting in an additional increase in the reaction rate. The reaction rate increases exponentially with temperature and with the square root of water vapor pressure, eventually increasing very rapidly until the reactants are exhausted.

CVDF-specific thermal runaway reaction analysis has been performed and is documented in HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report*, Annex B, "Cold Vacuum Drying Facility Final Safety Analysis Report," Section B3.4.2.5 (HNF 1999). The SSCs designed to prevent the MCO from runaway reaction are designated safety class because unmitigated consequences exceed the offsite release limit of 0.5 rem (0.005 Sv). However, the components of the Mark IA and Mark IV scrap basket copper shroud and copper subassembly need not be safety class because the copper serves to limit the fines placement and is used only to enhance thermal conductivity.

The respirable source term is based on the material at risk (MAR), the airborne release fraction (ARF), and the respirable fraction (RF). The MAR is the sum of the amount of particulate created before the release and the amount of particulate created during the release.

For a thermal runaway reaction at the CVDF, the MAR ranges from 80 kg to 185 kg of UO_2 . The ARF of all the radionuclides in the fuel, such as the fission products, may not be the same as that for the uranium particulate release under very high temperature conditions. An intermediate respirable release fraction of 5×10^{-3} has been chosen (Crowe et al. 1999). The unmitigated radiological offsite dose for a thermal runaway reaction event at the CVDF is above the offsite release limit of 0.5 rem (HNF 1999, Annex B).

Calculations in HNF-3553, Annex A, "Canister Storage Building Final Safety Analysis Report," Section A3.4.2.5 (HNF 1999), show that a thermal runaway fuel reaction accident is not credible at the CSB. This conclusion is based upon the analysis confirming the proof-of-dryness tests and characterization data collected to date (Lew et al. 1998). Because there is no release expected at the CSB, no source term has been estimated, and the inhalation dose consequences are zero.

11.4 HYDROGEN DEFLAGRATION OR DETONATION WITHIN A MULTI-CANISTER OVERPACK

Hydrogen gas can be generated within an MCO by two mechanisms: radiolytic decomposition of water (free water or water bound in aluminum hydroxide or uranium hydrates) or oxidation of uranium metal. If oxygen is present in an MCO, flammable or explosive mixtures could result.

There are two scenarios for hydrogen deflagration or detonation in an MCO. The first involves the possibility of hydrogen and oxygen generated by radiolysis accumulating in a sealed MCO. The creation of a flammable mixture is not expected to occur because the oxygen accumulated in the MCO for the 40-year interim storage period of an MCO will not reach the minimum oxygen concentration needed to support combustion. The accumulation of O₂ and H₂ is analyzed for the CSB in HNF-SD-SNF-TI-040, *MCO Internal Gas Composition and Pressure during Interim Storage* (Duncan and Plys 1998). The report shows that a combustible mixture will not be reached for an MCO that has been successfully processed at the CVDF. In addition, combustion of the hydrogen and oxygen produced by radiolysis, if it occurs, would not exceed the MCO design pressure after the cover cap is welded on because free water in the MCO is not sufficient. The maximum pressure during combustion is calculated to be around 175 lb/in² absolute and no more than 192 lb/in² absolute (Plys 1999b).

The second scenario involves air ingress into an MCO containing accumulated H₂. Air ingress into the MCO assumes a path into the MCO through which the gases from the explosion and any entrained particulate would vent.

The source term associated with either scenario is based upon the physical impact of the explosion to the MAR in the MCO. Because the quantity of MAR is distinctly different for MCOs at the CVDF and CSB locations, the source term for each location has been calculated separately. The bounding quantity of MAR at the CVDF location is 25 kg based on particulate generated during processing. At the CSB, the MAR includes the 25 kg from the CVDF plus an additional 9 kg formed during shipment to and storage of the MCO at the CSB for a total MAR of 34 kg of particulate. The calculations are documented in HNF-3553, Annex A, Section A3.4.2.3, and Annex B, Section B3.4.2.4 (HNF 1999).

At the CVDF, the unmitigated radiological offsite dose for this event is below offsite release limits while the unmitigated onsite dose for an anticipated event is above onsite risk evaluation guidelines. At CSB, the unmitigated radiological offsite dose for this event is below

offsite release limits while the unmitigated onsite dose also exceeds onsite evaluation guidelines. The long process tube, which enhances water removal and thereby minimizes the potential for hydrogen explosion, need only be designated as safety significant at best.

11.5 MULTI-CANISTER OVERPACK INTERNAL PRESSURIZATION EVENTS

Throughout the lifetime of an MCO, it is possible, though remotely, that internal pressure could increase, with the pressure eventually being relieved in an uncontrolled fashion. The SSCs designated to prevent or mitigate MCO overpressurization high-pressure blowdowns are designated safety class because unmitigated consequences exceed offsite release limits.

At the CVDF, the uranium–water chemical reaction generates hydrogen gas, which pressurizes the MCO. The steam or water vapor from the heated liquid water also increases MCO pressure (for an isolated MCO) and, more importantly, provides an oxidant for the exposed fuel in the scrap and fuel baskets. The increasing gas temperature in the MCO also helps pressurize the MCO. Blowdown at a pressure of 375 lb/in² gauge is assumed to occur about 96 hours after MCO isolation (HNF 1999, Annex B, Section B3.4.2.6). If the MCO were to be breached from high pressure, most of the gas and some suspended particulate would be carried out of the MCO in a very short time. The unmitigated radiological offsite dose for a high-pressure blowdown at the CVDF is above offsite release limits and onsite risk evaluation guidelines. Detailed calculations are presented in HNF-3553, Annex B, Section B3.4.2.6 (HNF 1999).

The safety-class components designated to prevent or mitigate the dose consequences of overpressurization accidents at the CVDF are the MCO shell assembly, the shield plug, the rupture disk, the locking and lifting ring and bolts, and the bottom plate.

MCOs are backfilled with helium to a prescribed pressure just before being sealed for shipment from the CVDF to the CSB. The pressure inside the MCO can increase because of the radiolytic decomposition of water and aluminum hydroxide, and the generation of hydrogen from the chemical reaction of water with fuel inside the MCO. Uncontrolled release of the MCO internal gas pressure, referred to as a pressurized gaseous release accident, results from failure of the pressure boundary confining the MCO gases. A pressurized gaseous release at the CSB would lead to the entrainment and release of fuel particulate from the MCO and the creation of a radiological hazard (HNF 1999, Annex A, Section A3.4.2.2).

The SSCs designated to prevent or mitigate MCO continuous releases at the CSB are designated safety significant because unmitigated consequences do not exceed offsite release limits. The unmitigated radiological dose to the onsite worker from the gaseous release accident is calculated to be greater than the onsite risk evaluation guidelines for anticipated events. Detailed calculations are presented in HNF-3553, Annex A, Section A3.4.2.2 (HNF 1999).

The safety-significant MCO components designated to prevent or mitigate the dose consequences of gaseous releases at the CSB are the seal, the 0.25-in orifice cover plates, the cover or orifice plate bolts, and the cover plates.

11.6 SUMMARY OF SAFETY-CLASS STRUCTURES, SYSTEMS, AND COMPONENTS

The safety-class and safety-significant MCO components that provide criticality control contingency features or are designated to prevent or mitigate potential accidents inherent to the MCO are summarized in Table 11-1.

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12.0 BASES FOR TECHNICAL CONTROLS AND LIMITS

12.1 INTRODUCTION

This chapter summarizes the necessary set of multi-canister overpack (MCO) key parameters for pressure, temperature, fuel mass, fuel type, scrap basket number, shipping window, free water limiting, and canister particulate resulting from design considerations or analyses that form the basis for developing the technical safety requirements (TSRs) needed to establish the safety envelope for MCO-related operations. Specific TSRs will be developed from this information and from facility-specific analyses, and will be published in the facility TSRs.

In DOE Order 5480.22, *Technical Safety Requirements*, TSRs are defined as safety limits, operating limits, surveillance requirements, and administrative controls. As stated in DOE Order 5480.22, TSRs shall define the operating limits and surveillance requirements, the bases thereof, safety boundaries, and management or administrative controls necessary to protect the health and safety of the public and to minimize the potential risk to workers from the uncontrolled release of radioactive or other hazardous materials and from radiation exposure due to inadvertent criticality. The key parameters in this section as well as other analyses in this MCO Topical Report and site-specific safety analysis reports (SARs) are source documents for TSRs.

12.2 KEY PARAMETERS

The key parameters in this chapter are necessary to ensure safe MCO performance. They are derived from analyses in this MCO Topical Report and in Spent Nuclear Fuel Project technical baseline documents, and they are summarized in Table 12-1.

12.2.1 Fuel Maximum Temperature

Eutectics or alloys with melting points lower than the separate constituents are a potential concern because of their structural impact and also because of the undesirability of the presence of molten material within the MCO. The metals in the MCO system are stainless steel (assumed to act as iron for eutectic interactions), copper, aluminum, uranium, and zirconium. The lowest eutectic temperature is for aluminum-uranium at 646 °C (1,195 °F); this is close to the melting point of aluminum of 660 °C (1,220 °F). The next higher eutectic temperature is for iron-uranium at 725 °C (1,336 °F). Whether the physical contact between the solid materials could lead to eutectic formation is doubtful. Nonetheless, since 646 °C (1,195 °F) is well beyond the temperatures the MCO can reasonably experience, there is merit in using 600 °C (1,112 °F) (the eutectic temperature with an additional decrement of ~50 °C [~90 °F] for conservatism) as a system limit to provide further protection for the MCO.

Table 12-1. Multi-Canister Overpack Key Parameters. (2 sheets)

Parameter	Applicability	Parameter value	Basis
Fuel maximum temperature	All times	600 °C (1,112 °F)	Lowest fuel structural material eutectic formation temperature
MCO pressure boundary maximum temperature (pressure or confinement boundary as defined in Section 1.3.2 of this MCO Topical Report)	All times	375 °C (707 °F)	ASME Boiler and Pressure Vessel Code
MCO external wall (shell, bottom plate, shield plug external surface, or cover cap external surface) temperature for CSB storage	Within storage tube in CSB	132 °C (270 °F)	Limit for concrete in CSB vault 1
Radial temperature gradients	All times	<100 °C (180 °F) between shell and shield plug center; <5 °C (9 °F) radially within MCO shell wall	Maintain MCO seal integrity
MCO internal pressure	When mechanically sealed When welded cover cap closure installed	150 lb/in ² gauge 450 lb/in ² gauge	Shield plug closure assembly pressure rating Design pressure rating with welded cover cap
Fuel mass maximum for Mark IV intact fuel, enveloping other fuel types and scrap	All times	≤6,835 kg	Crane loading limits must not be exceeded; maximum fuel decay heat assumptions and available MCO free volume assumptions depend on maximum fuel loading
Fuel type	All times	Mark IA fuel or scrap may not be placed in a Mark IV basket*	Critically safe geometry is maintained by basket design for the Mark IA fuel and by the MCO shell for Mark IV fuel
Scrap basket number and mass	All times	SPR fuel must not be mixed with any other type of fuel in an MCO No more than two scrap baskets will be placed in an MCO The mass of scrap in a Mark IA scrap basket will be 575 kg or less; the mass of scrap in a Mark IV scrap basket will be 980 kg or less	Criticality analyses for SPR fuel were performed assuming loading in Mark IA baskets with no other fuel types Assumptions of reactive fuel surface area and criticality analyses depend on number of scrap baskets and total mass of scrap per basket
Shipping windows	Flooded MCO shipment from K Basins to CVDF	24 hours	Actions required to vent pressure build-up

Table 12-1. Multi-Canister Overpack Key Parameters. (2 sheets)

Parameter	Applicability	Parameter value	Basis
Shipping windows	Dry shipment from CVDF to CSB	135 days	Actions required to avoid flammable gas build-up in the cask annulus
Process step limiting free water inventory	End of cold vacuum drying	≥ 40 °C (104 °F) hold at ≤ 12 torr for ≥ 8 hours with no scrap baskets, ≥ 20 hours with one scrap basket, or ≥ 28 hours with two scrap baskets	Thermal stability assumptions require < 200 g free water limit; process step at end of CVDF will ensure this limit
Fuel surface area	All times	(See Tables 4-2, 4-7)	Bounding calculations of surface areas from fuel geometry and fuel damage calculations
Fuel reaction rates and enhancement factors	All times	(See Table 4-2)	Literature values for uranium reaction rates in appropriate environments with additional multiplicative rate enhancement factors for conservatism
Particulate mass and water content	All times	(See Table 4-2, 4-4, 4-5)	Bounding values for adhering oxide films and particulate potentially loaded into MCOs, with associated chemically bound water content
Fuel decay heat	All times	(See Tables 4-2, 4-6)	Bounding calculated values from MCO fuel source inventory for various loadings

*The 12 length-code E Mark IA assemblies and the Mark IA assemblies stored in the K East Basin are exceptions. These long Mark IA assemblies will be loaded into a Mark IV basket. Criticality analyses (using two scrap baskets) have been performed and have yielded acceptable results.

CSB = Canister Storage Building.

CVDF = Cold Vacuum Drying Facility.

MCO = multi-canister overpack.

SPR = single pass reactor.

12.2.2 Multi-Canister Overpack Pressure Boundary Maximum Temperature

The maximum MCO pressure boundary temperature, if based solely on the ASME *Boiler and Pressure Vessel Code* (ASME 1998), is 427 °C (800 °F). This is appropriate based on the MCO boundary, which is constructed largely of 304L stainless steel in accordance with the Boiler and Pressure Vessel Code, Section III, Subsection NB (ASME 1998). Applicable components are the MCO shell, shield plug, MCO collar, MCO bottom baseplate, port covers, MCO cover cap, and weldments. Selection of 304L stainless steel effectively limits the maximum temperature to 427 °C (800 °F) since L grades exhibit low strength in higher temperature ranges. The Boiler and Pressure Vessel Code (ASME 1998) case N 47 extends the rules of Subsection NB (ASME 1998) to temperatures above 427 °C (800 °F), but this case does not permit construction using 304L stainless steel. The 427 °C (800 °F) temperature is reduced by 50 °C as an additional factor of conservatism to result in a system limit of 375 °C (707 °F). This limit is only applicable in reference to the ASME Boiler and Pressure Vessel Code.

12.2.3 Multi-Canister Overpack External Wall Temperature for Canister Storage Building Storage

The most restrictive temperature limit is 132 °C (270 °F) at the MCO external wall (shell, bottom plate, shield plug external surface, or cover cap external surface). This is based on the concrete temperature design requirement of the Canister Storage Building (Goldmann 2000b). ACI 349-85, *Code Requirements for Nuclear Safety Related Concrete Structures*, Appendix A.4, provides limitations for concrete temperatures during normal operations or any long-term period. The concrete temperature shall not exceed 150 °F.

12.2.4 Radial Temperature Gradient

The maximum temperature difference between the MCO shell and the shield plug must be less than or equal to 100 °C (180 °F) between the outside of the shell and the center of the shield plug, and less than or equal to 5 °C (9 °F) radially within the MCO shell wall. This criterion is from Section 4.9.2 of HNF-S-0426, *Performance Specification for Spent Nuclear Fuel Multi-Canister Overpack* (Goldmann 2000b). The parameter is required to limit thermal expansion and preserve the integrity of the MCO mechanical seal.

12.2.5 Multi-Canister Overpack Internal Pressure

The MCO pressure limit is based on the specific design and materials selected and is dependent on the mode of sealing. When mechanically sealed, the design pressure for the shield plug closure assembly is 150 lb/in² gauge (Goldmann 2000b). When a closure cap is welded over the shield plug, the design pressure for the MCO shell, bottom plate, and cover cap is 450 lb/in² gauge (Goldmann 2000b).

12.2.6 Fuel Mass Maximum

The total mass of the cask and MCO combination to be lifted by the crane must be less than the K Basins' 32-ton crane limit, as well as less than subsequent facilities' crane limits. The fuel mass portion of the total mass is based on Table 4-5 of this MCO Topical Report, which states that a maximum fuel load of 270 Mark IV assemblies contains 6,835 kg (15,071 lb). This is derived from the maximum weight of five baskets of Mark IV fuel plus the maximum amount of particulate loaded with the fuel. On a single basket basis, this is 1,367 kg (3,014 lb). This is consistent with the weights given in Table 3-6 of this MCO Topical Report with the additional weight of the fuel particulate from Table 4-5. This limit ensures that available free volumes given in Tables 4-3 and 4-5 are obtained in the MCO for pressure accommodation and also ensures that the total decay heat from the uranium fuel mass remains below the 776 W maximum assumed in thermal analyses.

12.2.7 Fuel Type

The MCO is critically safe by reliance on basket design for Mark IA fuel. During accident conditions the Mark IA fuel and scrap baskets are relied upon to retain structural integrity, meeting service Level D requirements under ASME Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME 1998), as stated in Section 4.12.1 of HNF-S-0426 (Goldmann 2000b). The Mark IV fuel is lower in enrichment and therefore the Mark IV fuel and scrap baskets do not need to maintain structural integrity to remain critically safe; the MCO itself is relied upon to retain structural integrity in accident conditions, as stated in Section 4.12.2 of HNF-S-0426 (Goldmann 2000b).

Because of the difference in fissile enrichment between the two fuel types, they are normally segregated and not mixed within an MCO. Mixing is allowed for the 26.1-in.-long Mark IA assemblies in the K West Basin and for the small amount of Mark IA material in the K East Basin. Analyses performed for single pass reactor fuel also require that it be loaded into a single MCO and not mixed with other fuel types (Sellers 1998).

There is a limited amount of 26-in. Mark IA fuel that will require additional processing to ensure fitting within the Mark IV basket. This processing method is to be determined.

12.2.8 Scrap Basket Number and Mass

The criticality analyses for the MCO are applicable to a maximum of two scrap baskets per MCO, with one at the top position and one at the bottom position of the MCO (Kessler 2000). Assumptions of maximum reactive surface within an MCO depend on no more than two scrap baskets per MCO (Reilly 1998). The maximum number of scrap baskets and their position is specified in Sections 3.2.1 and 3.2.2 of HNF-SD-SNF-OCD-001 (Pajunen 2000).

The total mass of scrap allowed to be loaded into a Mark IA scrap basket is limited to 575 kg. The total amount of scrap allowed to be loaded into a Mark IV scrap basket is limited to 980 kg. These limits are necessary to remain within limits established for the criticality analyses (Chapter 6.0, Section 6.1.1).

12.2.9 Shipping Windows

The design shipping window for transportation from the K Basins to the Cold Vacuum Drying Facility (CVDF) is 24 hours (Smith 2000). The available time before cask pressure relief is required is 32 hours for a safety basis MCO, from Section 4.4.3.1 of this MCO Topical Report. No excursions were predicted by the thermal analyses.

The design shipping window for transportation from the CVDF to the Canister Storage Building is 135 days (Smith 2000). A safety basis MCO will not undergo a thermal excursion and will not require pressure relief, from Section 4.4.3.2 of this MCO Topical Report. Action may be required at the end of the shipping window to mitigate flammable gas buildup in the cask resulting from MCO leakage.

12.2.10 Process Step Limiting Free Water Inventory

Thermal analyses for MCO processing and storage after cold vacuum drying assume that less than 200 g of free water is available to react with fuel. This limit is assured by a processing step at the end of cold vacuum drying, described in Section 3.4.1 of HNF-SD-SNF-OCD-001 (Pajunen 2000), consisting of a hold period of at least 28 hours, for an MCO with two scrap baskets, at a temperature of at least 40 °C (104 °F) and at a pressure less than or equal to 12 torr. A 20 hour or greater hold will be required for an MCO with one scrap basket and 8 hours or greater for an MCO with no scrap baskets.

12.2.11 Fuel Parameters Necessary for Thermal Analysis

Values are required for several parameters describing the physical and chemical properties of the fuel in order to perform the analyses that confirm the thermal stability of the loaded MCO:

- Fuel surface area per MCO and per basket
- Recommended fuel reaction rates and enhancement factors
- Particulate mass and water content
- Decay heat per MCO.

The values for these parameters and the bases for their selection are given in HNF-SD-SNF-TI-015, *Spent Nuclear Fuel Project Technical Databook* (Reilly 1998). They also are described in Chapter 4.0, Section 4.2.1.2, and provided in full in Tables 4-2, 4-5, 4-6, and 4-7.

12.3 DESIGN FEATURES

The following design features are of special importance to each of the physical barriers, maintenance of safety margins, and establishment of TSRs for the MCO:

- Pressure boundaries (from Chapter 3.0, Section 3.1.3)
 - Shell assembly and cover cap — 450 lb/in² gauge at 132 °C (270 °F)
 - Shield plug closure assembly — 150 lb/in² gauge at 132 °C (270 °F)
- Criticality control (from Chapter 6.0, Section 6.1.1, and Chapter 3.0, Section 3.1.10.4)
 - Shell inside diameter (22.985 in. nominal)
 - Central insert pipe, nominal 6.625 in. outer diameter, at longitudinal centerline of MCO (required for Mark IA fuel loadings only)
 - Basket baseplate (required for Mark IA fuel loadings only)
- Radiological shielding (from Chapter 5.0, Section 5.3.1)

Modeling for shielding is based on MCO characteristics described in detail in Chapter 1.0, Section 1.2, and summarized as follows (only MCO characteristics are shown below; reflector properties assumed for shielding calculations are not given)

- MCO shell is made of stainless steel
- MCO shield plug is made of stainless steel
- Inside diameter of the MCO is 23 in. and inside height is 148 in. to the bottom of the MCO shield plug
- MCO shield plug is 10.5 in. thick
- MCO sidewall is 0.5 in. thick and reflector sidewall is 8 in. thick for a total thickness of 8.5 in.
- MCO bottom is 2.0 in. thick.

12.4 ADMINISTRATIVE CONTROLS DURING DESIGN

Administrative control includes control of design changes, compliance with process parameters, and quality assurance controls.

- The use of the Engineering Change Notice process ensures MCO design configuration control and the thorough evaluation of changes and their impacts on all operational phases.
- The use of technical maintenance and operations procedures ensures that process parameters are controlled within defined limits and that specified environmental conditions are maintained.

All design changes to the MCO will be assessed by the Engineering Change Notification process (HNF-PRO-440). The MCO design authority will review all design changes for impact to safety and environmental requirements and notify engineers and organizations as appropriate. Impact to safety and/or environmental requirements will require review and sign-off of the Engineering Change Notification by individuals independent from the design authority and the original work.

U.S. Department of Energy (DOE) approval will be required for changes to functional design criteria, SARs, safety analysis reports for packaging, and the documentation of a positive unreviewed safety question, or other documents as required by HNF-PRO-233, *Review and Approval of Documents*, but specific design changes that do not affect the safety basis, such as routine Engineering Change Notification or Design Change Notification forms, will not require DOE approval. The MCO Topical Report will be part of the authorization basis for the SNF Project, therefore a change to this MCO Topical Report will require unreviewed safety question screening and subsequent appropriate actions.

13.0 QUALITY ASSURANCE

13.1 INTRODUCTION

The Spent Nuclear Fuel (SNF) Project quality assurance program provides assurance that the design, procurement, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification, including documentation requirements for activities, conform to regulatory and contractual requirements. This chapter summarizes the SNF Project quality assurance program for activities associated with the multi-canister overpack (MCO) and baskets. It also provides references to documents containing additional details of the SNF Project quality program.

The extent to which quality requirements are applied to the SNF Project is based on a graded approach reflecting safety significance and/or safety implications. Quality-affecting activities performed by organizations that provide equipment, services, or support to the SNF Project also are described in this chapter.

Adherence to the quality assurance program will ensure that

- U.S. Department of Energy (DOE) missions and objectives are effectively accomplished
- Products and services are safe, reliable, and meet or exceed the requirements and expectations of the user
- Any hazards to the public, to Hanford Site and facility workers, and to the environment are minimized.

13.2 REQUIREMENTS

The requirements that form the basis for the quality assurance program are found in HNF-3552, *Spent Nuclear Fuel Project Execution Plan* (Williams 1998) and in QAPP-OCRWM-001, *Quality Assurance Program Plan for Implementation of the OCRWM QARD for the Spent Nuclear Fuel Project* (Ruth 1999). These documents provide the quality assurance requirements for federal repository acceptance of SNF and satisfy U.S. Nuclear Regulatory Commission equivalency requirements. Any additional requirements applicable to specific SNF Project facilities are identified in the SNF Project Final Safety Analysis Reports (FSARs).

HNF-SP-1228, *Quality Assurance Program Implementation Plan for Nuclear Facilities* (Byers 1997), and HNF-MP-599, *Project Hanford Quality Assurance Program Description*, provide for the implementation of Title 10, *Code of Federal Regulations*, Part 830, "Nuclear Safety Management," Section 830.120, "Quality Assurance Requirements" (10 CFR 830).

The applicable quality assurance program plan (QAPP) and the Office of Civilian Radioactive Waste Management (OCRWM) requirements matrix maintained by the Fluor Hanford Quality Assurance organization identify the management processes and procedures for implementing DOE/RW-0333P, *Quality Assurance Requirements and Description* (QARD) (DOE/RW 1999), HNF-MP-599, and project-specific quality assurance requirements. Changes that reduce requirements relative to these requirements shall not occur before approval is obtained from the U.S. Department of Energy, Richland Operations Office.

13.3 QUALITY ASSURANCE PROGRAM ORGANIZATION

13.3.1 Program

The governing QAPP is the overall quality assurance program and contains criteria, requirements, and implementation documents for project activities. Applicable QAPPs are maintained to ensure compliance to requirements specified in 10 CFR 830.120; Title 10, *Code of Federal Regulations*, Part 71, "Packaging and Transportation of Radioactive Material," Subpart H, "Quality Assurance" (10 CFR 71); Title 10, *Code of Federal Regulations*, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Subpart G, "Quality Assurance" (10 CFR 72); and DOE O 414.1, *Quality Assurance*, as implemented by HNF-MP-599.

The QAPP quality assurance policy statement includes the following.

- Compliance with the provisions of a QAPP is mandatory.
- The SNF Project has committed to implement an effective quality assurance program that is in compliance with the applicable DOE and federal regulatory requirements as presented in Section 13.2.
- Quality assurance controls are written into SNF Project implementing procedures that are used to control work processes.
- Personnel shall be appropriately trained to the procedures they will use prior to the start of the work.
- No work subject to the requirements of the governing documents shall be started prior to the development, review, approval, and issuance of the appropriate work documents or procedures.
- Achievement of quality is a line responsibility, where each performer is accountable for the quality of the work assigned. Line organizations and line management are responsible for the effectiveness of the quality program for SNF Project.

The QAPP assigns prime responsibility for the implementation of the quality program to the SNF Project management and the managers of those organizations that support the SNF Project.

The degree of control over activities affecting the quality of items and services ensures conformance to the approved SNF Project design and is commensurate with the importance to safety. SNF Project personnel have the authority to stop work in accordance with approved procedures for unsafe situations and to control further operation until the conditions that created the unsafe conditions are corrected. Stop work actions are immediately communicated to project management.

Suppliers and subcontractors providing services for the SNF Project will be required to implement a quality assurance program consistent with the requirements specified in the SNF Project procurement documents. Quality assurance programs used by subcontractors providing safety-related supplies and services are submitted to the SNF Project for review and approval.

Readiness reviews are performed in accordance with approved procedures for determining compliance of the SNF Project and activities with applicable requirements.

Compliance with the QAPP is achieved by implementing written procedures identified in the QAPP within a baseline implementation matrix, supplemental baseline implementation matrix, and/or quality assurance program index. A QAPP presents the overall structure of the hierarchy of documents to meet the quality program requirements.

Application of procedures to project activities is described in the QAPP. Procedures identified in a baseline implementation matrix are applied to all SNF Project elements unless exceptions are justified. A supplemental baseline implementation matrix identifies additional procedures to implement SNF subproject-specific requirements. Requirements have been identified that shall be applied to structures, systems, and components (SSCs) that must also comply with 10 CFR 71, Subpart H, and/or 10 CFR 72, Subpart G. Implementing procedures for these requirements are presented in a quality assurance program index. A quality assurance program index does not apply to SNF Project work being performed that is not applicable to these standards. The SNF Project will ensure that the quality assurance program requirements will remain in effect during the lifetime of the SNF Project.

13.3.2 Personnel Training and Qualification

Personnel performing activities affecting quality involved with design, scientific investigation, software development, inspection, and nondestructive examination and testing are trained and qualified to perform assigned tasks in accordance with requirements specified in DOE/RW-0333P, paragraph 2.12.12B (DOE/RW 1999). This includes knowledge of the work processes, tools, equipment, and requirements. The training and qualification program will provide for the development of personnel proficiency commensurate with the scope, complexity, and nature of an assigned activity. Training and indoctrination needs are identified and

documented by management. Records shall be maintained to enable verification of personnel qualification and completion of required indoctrination and training.

SNF Project management, and managers of those organizations that support the SNF Project, are responsible for ensuring that their staffs are sufficiently trained to perform assigned tasks in a manner that minimizes (1) risk to the worker performing the task, coworkers, and the public; (2) negative impacts to the environment; and (3) risk of damage to equipment. Training and qualifications for specified job functions are based on analysis of the specified duties and tasks associated with the functions. Job functional descriptions are developed that describe minimum requirements for education, experience, and when required, physical condition, and certification. Management is responsible to verify qualification of personnel before assigning them to do work. Inspection and nondestructive examination positions are certified in accordance with written procedures that are based on ASME NQA-1-1994, *Quality Assurance Requirements for Nuclear Facility Applications* (ASME 1994), ASNT-SNT-TC-1A-1992, *Recommended Practice for Non-Destructive Testing Personnel Qualification and Certification* (ASNT 1992), or other standards applicable to the work performed.

Qualification and training requirements are periodically reviewed to ensure that they continue to reflect the current systems, procedures, and policies applicable to each position. Continuing training is provided to ensure that job proficiency is maintained. Training is provided by instructors having the technical and instructional skills necessary to provide the training in an effective manner. Training program effectiveness and efficiency is determined by training management through feedback from instructors, students, students' managers, and periodic reviews. Improvements are addressed to maintain or increase proficiency based on existing or new requirements.

The SNF Project personnel training and qualifications process meets the requirements of Part 2, Section 2, "Personnel Training and Qualifications," of HNF-MP-599. Implementing procedures for personnel selection, indoctrination, training, and qualification are presented in the QAPP.

13.4 QUALITY IMPROVEMENT

The objective of quality improvement is to detect and correct problems adversely impacting quality and to continuously improve the quality of items and work processes. The basis of the approach to quality improvement is that (1) work activities can be planned, performed, assessed, and improved; and (2) lessons learned from this process can be used when planning subsequent activities to preclude recurrence of problems.

Under the quality improvement program, organizations have the authority to identify quality problems and to initiate, recommend, or provide solutions through designated channels. Conditions adverse to quality (including failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances) are promptly identified and corrected. Management systems for quality improvement include problem identification, screening,

evaluation, reporting and resolution; nonconformance identification, control and tracking; corrective action process; and tracking problems to closure.

Item characteristics, process implementation, and other quality-related information are reviewed and data analyzed to identify items, services, and processes needing improvement. This includes an assessment of risk. Items, services, and processes that do not meet established quality requirements are identified, controlled, and corrected promptly in accordance with the significance of the problem and work. Verification of corrective action completion is also conducted.

A problem identification process is used to identify problems that are then evaluated for an unreviewed safety question and *Price-Anderson Amendments Act of 1988* violation subject to Title 10, *Code of Federal Regulations*, Part 820, "Procedural Rules for DOE Nuclear Activities" (10 CFR 820). Correction includes identifying the cause of the problem, assessing risk, determining corrective action for the problem, performing problem closure, and initiating any additional corrective actions to preclude recurrence.

The SNF Project will establish a process to review U.S. Nuclear Regulatory Commission Inspection and Enforcement Bulletins and Notices communicating identified defective equipment in the commercial nuclear power industry. Appropriate actions will be taken on similar safety-class equipment in the SNF Project.

Periodic inspection of SSCs and equipment is performed to determine whether deterioration is taking place and to identify and address obsolescence that threatens personnel safety or facility preservation. Performance indicators, item characteristics, process implementation, corrective actions, assessments, and other quality-related information are reviewed and data analyzed to identify items, services, and processes needing improvement. When appropriate, these improvement evaluations include use of tools such as root cause analysis and lessons learned. Problem identification processes and nonconformances are periodically analyzed to identify quality trends.

The SNF Project quality improvement process meets the requirements of Part 2, Section 3, "Quality Improvement," of HNF-MP-599. Implementing procedures for quality improvement requirements are presented in the QAPP.

13.5 DOCUMENTS AND RECORDS

The documents and records program establishes requirements for control of the preparation, review, approval, issuance, use, and revision of documents that prescribe processes, specify requirements, or establish design. Controlled documents, including revisions, are reviewed for adequacy, completeness, and correctness before approval by the organization that originally reviewed and approved the document, or by a designated alternate organization with comparable technical competence and capability. After approval, controlled documents are released to specified users to ensure the latest approved revisions are available to personnel for

use at the location where the work is being performed. Compliance to these controlled documents is required.

Records are specified, prepared, reviewed, approved, maintained, and stored in accordance with approved procedures and instructions. Lifetime and nonpermanent records classification and records retention requirements are specified in the procedures that govern generation of the records. These procedures comply with DOE/RW-0333P, Section 17.0 (DOE/RW 1999). MCO- and basket-related design, fabrication, and storage information are considered lifetime quality assurance records.

Records are to be legible, identifiable, accurate, complete, protected, and retrievable. Maintenance of records includes provisions for correction, replacement, retention, preservation, traceability, and accountability. Computer hardware and software used to maintain, index, store, or access records are maintained and controlled to ensure accountability, reproducibility, and protection from loss.

The SNF Project documents and records process meets the requirements of Part 2, Section 4, "Documents and Records," of HNF-MP-599. Implementing procedures for document and record requirements are presented in the QAPP.

13.6 QUALITY ASSURANCE PERFORMANCE

This section provides an overview of the process that the SNF Project uses to ensure that the performed work meets requirements. Work processes, design, procurement, and testing processes are covered in Sections 13.6.1 through 13.6.4.

13.6.1 Work Processes

Work processes include, but are not limited to, activities involving design, analysis, fabrication, procurement, handling, shipping, storing, cleaning, assembly, inspection, and testing. Work on the SNF Project is planned, authorized, and performed under controlled conditions in accordance with approved technical standards and administrative controls using approved procedures, instructions, plans, or other control documentation commensurate with the complexity and risk posed by the work to be performed. Such procedures, instructions, plans, or other control documentation contain or reference all of the necessary administrative and technical requirements, including the sequence of actions and interactions, as required to ensure that activities are properly performed and meet acceptance criteria or other requirements.

Work process documents are developed, reviewed, and approved by personnel technically knowledgeable of the work. Work process documents are readily accessible to the worker. Supporting documentation for work activities is reviewed to ensure that the desired quality is being maintained and to identify areas for improvement. Use of such controls is to ensure that process parameters are controlled within defined limits and that specified environmental

conditions are maintained. Work completion is documented and appropriate records maintained. Special processes (e.g., welding, heat treating, nondestructive testing) are performed by major subcontractors or other qualified subcontractors in accordance with approved processes. Specific requirements are established in procurement documents.

Identification is maintained on items, or in documents traceable to them, in a manner that ensures identification is established and maintained to ensure control and maintenance of items for manufacture or receipt through delivery, installation, or use. Items may include hardware, samples, or data. This item identification and administrative control is used to prevent the use of incorrect or defective items and to maintain traceability for items, as required by specifications, codes, and standards. Handling, marking, storing, packaging, shipping, cleaning, and preserving materials and other items is controlled to prevent damage, loss, or deterioration. Marking and labeling of items is maintained throughout the processes of packaging, shipping, handling, and storage. Any status indicators will not be detrimental to the item.

Calibration and maintenance of equipment used for process monitoring and data collection are conducted in accordance with approved procedures. Computer software used in applications important to environmental, safety, health, and quality aspects shall be subject to appropriate controls throughout the software's life cycle.

The fabricator is required to have an acceptable quality assurance program that shall be in effect throughout the fabrication, assembly, testing, inspection, packaging, and shipping of the MCO, including receipt inspection by the buyer. The fabricator's quality assurance program shall meet the applicable requirements of the American Society of Mechanical Engineer's (ASME) *Boiler and Pressure Vessel Code*, Section III, Division 1, NCA-4000 (ASME 1998), as specified by HNF-S-0453, *Specification for Multi-Canister Overpack Fabrication* (Goldmann 1998), and 10 CFR 830, as specified in the procurement statement of work. The fabricator's quality assurance program shall be accredited by ASME and must be accepted by the buyer before any work is performed, including procurement of materials.

Quality records shall be traceable to the items, components, or activities to which they apply through use of unique identifiers, as well as governing purchase order numbers. The fabrication data package shall include as-built drawings, nonconformance reports and/or deficiency reports including satisfactory resolutions, certified materials test reports, hydrostatic test reports, nondestructive examination reports, and inspection reports. The data package shall be traceable to each MCO and shall provide a clear history of the materials, processes, inspections, tests, nonconformance or deficiency reports, and repairs, and verification that required personnel and processes were qualified and that appropriate and approved procedures were used and complied with. These documents shall be controlled as lifetime quality records and shall be stored, controlled, traceable, and retrievable. The fabricator's cleaning, handling, and shipping plans and/or procedures shall be reviewed and approved by the buyer.

Measures to control the handling, storage, shipping, cleaning, and preserving of materials and equipment to prevent damage and deterioration are contained in Appendix 17, "Warehouse Plan," of HNF-SD-SNF-DR-003, *Multi-Canister Overpack Design Report* (Goldmann 2000a).

The quality assurance requirements to be followed during all stages of transporting an MCO loaded with fuel are contained in HNF-SD-TP-SARP-017, *Safety Analysis Report for Packaging, Onsite, Multi-Canister Overpack Cask*, the safety analysis report for packaging (SARP) (Smith 2000). Specific instruction regarding when the SARP requirements apply is contained in the SARP.

The SNF Project work process meets the requirements of Part 2, Section 5, "Work Processes," of HNF-MP-599. Implementing procedures for work processes are presented in the QAPP.

13.6.2 Design

Sound engineering and scientific principles, codes, standards, and practices for the assurance of technical quality are identified and incorporated into the design of new or replacement items and system design requirements. The design, including design changes, incorporates applicable design requirements, including design bases, functional and performance requirements, regulatory requirements, codes, standards, and environmental conditions, and interfaces with new or existing items.

The SNF Project design process meets the requirements of Part 2, Section 6, "Design," of HNF-MP-599. Implementing procedures for design are presented in the QAPP.

13.6.2.1 Design Inputs. Design, including design changes, incorporates design inputs consisting of controlled requirements, customer expectations, and applicable design requirements, including design bases, functional and performance requirements, federal codes, DOE orders and standards, conceptual design criteria, national standards, specifications, drawings, environmental conditions, health and safety considerations, expected life cycle, reliability requirements, and interfaces with new or existing items. Design inputs are identified and documented and their selection reviewed and approved by the responsible design organization to ensure that sound engineering and/or scientific principles and appropriate standards are being used.

13.6.2.2 Design Process. Appropriate quality standards are identified and documented as design input and their selection reviewed and approved. Design methods, materials, parts, equipment, and processes that are essential to the functions of the items affecting quality are selected and independently reviewed for suitability of application. Identified deviations from quality standards are controlled in accordance with procedures or instructions.

Independent design reviews are performed, documented, and comments resolved, and the results of the reviews are identifiable, stored, controlled, and retrievable. These independent reviews ensure that the requirements of HNF-S-0426, *Performance Specification for Spent Nuclear Fuel Multi-Canister Overpack* (Goldmann 2000b), have been met and that the design is in accordance with the applicable portions of 10 CFR 830 and the ASME *Boiler and Pressure Vessel Code*, Section III, Subsections NB and NG, for the baskets (ASME 1998). When a test

program is used to verify the adequacy of the design in lieu of other design verification methods, the qualification testing of the item shall include testing under the most adverse design conditions.

The design agent must execute quality assurance programs that, in addition to satisfying the requirements previously stated, provide the following assurances.

- Performance requirements and design criteria are established, documented, and clearly understood.
- Studies, analyses, and design decisions are fully documented.
- Performance requirements and design criteria are met.
- Design is complete, adequate, and properly documented.
- Traceability to the requirements of the contract technical specification is maintained.

The final design of the SNF Project facilities is related to the design bases input by documentation of sufficient detail to permit design verification and by identification of assemblies or components that are part of the item being designed. Design documents are adequate to support facility design evolution, construction, operation, and deactivation. The documentation references include the applicable codes, standards, and practices, in addition to the applicable regulatory safety requirements.

Design interfaces are identified and controlled and design efforts coordinated among the participating organizations. Interface controls are documented and include the assignment of responsibility and the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces. Design information transmitted across organizational interfaces is documented and controlled. Transmittals identify the status of the design information or document provided and, where necessary, identify incomplete items that require further evaluation, review, or approval.

Use of commercial-grade items is accomplished in accordance with documented processes using recognized industry standards.

13.6.2.3 Design Verification and Validation. Design verification or validation of the adequacy of design products is accomplished by technically knowledgeable persons other than those who performed the work.

Design adequacy is verified in accordance with approved procedures by individual or interdisciplinary design reviews, alternate calculations to verify the correctness of the original design calculations, or qualification testing to demonstrate adequacy of performance under conditions that simulate the most adverse design conditions. The extent of verification is based on the complexity, risk, and uniqueness of the design.

Design verification is performed in a timely manner and identified design errors corrected before approval and implementation of the design. Design verification, for the level of design activity accomplished, is performed before release for procurement, manufacture, construction, or release to another organization for use in other design activities except in those cases where this timing cannot be met, such as when insufficient data exist. In those cases, the unverified portion of the design is identified and controlled. In all cases, design verification is completed before relying upon the item to perform its function.

Engineering documents (e.g., drawings, specifications, design analyses, system descriptions, engineering studies, technical reports) are verified in accordance with approved procedures and instructions.

13.6.2.4 Design Changes. Design changes, including field changes, and nonconforming items dispositioned "use-as-is" or "repair" are justified and subject to design control measures commensurate to those applied to the original design. These measures include assurance that the design analyses establishing the safety basis for the SSCs are still valid. Verification and review of design changes are performed to the same level as performed for the original design. As-built changes are documented and verified. Design of temporary modifications receives the same level of control as the original design for permanent modifications.

13.6.2.5 Design Documentation and Records. The final design is related to the design input by documentation of sufficient detail to permit design verification, and by identification of assemblies and/or components that are part of the item being designed. Design documentation is adequate to support facility design, construction, and operation. Design output documents include drawings, specifications, test and inspection plans, and maintenance requirements. As-built drawings and shop drawings are maintained to show actual plant configuration. The administrative process clearly indicates responsibilities for design output document activities including marking-up and updating during construction and operation of SNF Project activities.

13.6.2.6 Computer Software. Computer programs used for design analysis are verified to show that they produce correct solutions for the encoded mathematical model within defined limits for each parameter employed. Computer programs also are verified to show that the encoded mathematical model produces a valid solution over the range of applications to the physical problem associated with the particular application. Computer programs are controlled to ensure that changes are documented and approved by appropriate personnel; control requirements include development, acquisition, use, modification, and configuration management of software used in computer systems. The requirements of DOE/RW-0333P, Supplements I and V (DOE/RW 1999), are applicable to MCO and basket QARD-related activities.

13.6.3 Procurement

Procurement of items and services by the SNF Project, or by its subcontractors, is documented and controlled to ensure that regulatory requirements, design, and other necessary quality requirements are included or referenced in the documents used for procurement of items

and services, and that they perform as specified. Procurement documents are controlled to ensure that applicable requirements, design bases, testing and inspection requirements, acceptance criteria, and other requirements necessary to ensure adequate quality are included or referenced in documents for procurement of items and services. Critical parameters and requirements are specified including submittal, product-related documentation, nonconformance requirements, administrative documentation, personnel or materials qualifications, tests, inspections, and reviews.

A graded approach is used when establishing the stringency of procurement requirements to ensure they are commensurate with the importance of the purchased item or service. Applicable quality assurance requirements are applied to suppliers through the procurement documents issued to these suppliers. Changes also are communicated to suppliers in the procurement documentation. Such changes are evaluated in the same manner and use the same criteria as the original documents. The procurement documents also specify the right of access to suppliers' (including subtier suppliers') facilities for surveillance of work activities, inspection of facilities and programs, review of plans and program reports, processing of change information, and review of disposition of nonconformances.

Controls are established to ensure that purchased items and services conform to the procurement documents. These controls include provisions for source evaluation and selection, objective evidence of inspection at the contractor or subcontractor source, examination of items or services upon delivery, and assessments. Requirements that the supplier report defects and noncompliances of safety-class and safety-significant items and services are incorporated into procurement documents. Nonconformance procedures are in place for disposition of items or services that do not meet procurement requirements.

Documentary evidence that an item conforms to code, regulation, or contract procurement requirements is completed before installation or use of the item. Methods established for the acceptance of an item furnished by suppliers may consist of one or more of the following:

- Supplier certification and release (Certificate of Conformance)
- Source verification or inspection
- Receiving inspection
- Acceptance testing.

Supplier-generated documents are controlled in accordance with the requirements in Section 13.5.

An assessment to determine the effectiveness of the control of quality by the supplier is conducted, evaluated, and documented before selection and periodically during supplier performance at intervals consistent with the importance, complexity, and quantity of the items or services. The evaluation and selection of procurement sources is based on specified criteria. The evaluation includes one or more of the following:

- Evaluation of the supplier's quality history of providing an identical or similar product that performs satisfactorily in actual use
- Review of the supplier's current qualitative and quantitative information that can be objectively evaluated
- Direct evaluation of the supplier's facilities, personnel, and quality assurance program implementation to determine the technical and quality capability of that supplier.

Assurance is obtained that approved suppliers can continue to provide acceptable items and services based on a documented evaluation of their past performance. Suppliers are periodically reevaluated and either retained on the basis of continued satisfactory performance, or removed from a list of acceptable suppliers. This performance history evaluation includes the following items:

- Evaluations of the supplier's nonconformance report history relative to received items or services
- Communications with the purchasing organization to determine whether any contractual problems have been encountered
- Communications with the supplier to determine whether any changes have occurred to their quality assurance program since their initial acceptance.

Safety-class and safety-significant materials that must meet the requirements of the ASME Boiler and Pressure Vessel Code (ASME 1998) shall be procured from suppliers or manufacturers whose quality assurance programs have been evaluated and found to be acceptable for compliance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NCA-3800 (ASME 1998). The QAPP requires that safety-class and safety-significant items and services be procured from suppliers, vendors, or manufacturers who possess evaluated and approved quality assurance programs. Appropriate ASME code cases may be used with buyer's agreement prior to the initiation of material procurement.

Those safety-significant materials that are not designated as having to meet the requirements of the ASME Boiler and Pressure Vessel Code (ASME 1998) shall be procured from suppliers or manufacturers whose quality assurance programs have been evaluated and found to be acceptable for compliance with the applicable requirements of the procurement statement of work.

Procurement of general services materials shall meet the quality requirements, if any, and the technical requirements of the procurement documents.

The SNF Project procurement process meets the requirements of Part 2, Section 7, "Procurement," of HNF-MP-599. Implementing procedures for procurement are presented in the QAPP.

13.6.4 Inspection and Acceptance Testing

Inspection and acceptance testing of specified items, services, and processes are performed in accordance with established acceptance and performance criteria. Inspection and acceptance criteria are derived from engineering design documents, supplier information, construction procedures, and maintenance procedures. Inspection, surveillance, and testing of items, services, and processes that have the potential to affect quality during procurement, construction, repair, modification, maintenance, and installation are subject to these requirements on a graded approach based on safety class.

Hold points (e.g., items or activities where inspection is mandatory), witness points, verification points, methods, acceptance criteria, checklists are established, documented, and implemented to ensure required inspections are performed. Test requirements and acceptance criteria are identified, documented, and approved. Test results are documented and their conformance with acceptance criteria evaluated by a responsible authority to ensure that test requirements have been satisfied.

The inspection and testing process establishes the system by which the inspection and testing status of items is controlled to ensure that items that have not passed the required inspections and tests are not inadvertently installed, used, or operated. Status of inspection and test activities is identified either on the item, by markings such as stamps, tags, labels, or routing cards, or in documents traceable to the items, including heat number, part number, serial number, or other appropriate means throughout fabrication, installation, and use.

Controls provide for the identification of items that have satisfactorily passed required inspections and tests where necessary to preclude inadvertent bypassing of the inspection and testing.

Inspections and tests are performed in accordance with approved procedures, instructions, or inspection plans by qualified personnel to demonstrate that the SSCs will perform satisfactorily in service. This documentation contains the following information:

- References to applicable documents such as drawings, specifications, and procedures
- Type of inspection and testing to be performed
- Mandatory hold points, when required
- Characteristics to be inspected and tested
- Qualification and individuals or groups responsible for performing the inspection and testing
- Acceptance and rejection criteria (explicit or by reference) obtained from specifications, drawings, supplier instructions, and standards

- Suitable environmental conditions
- Description of the inspection and testing method and equipment to be used, or reference to an appropriate procedure (including adequate testing instrumentation, equipment, calibration requirements, and environmental conditions)
- Required safety measures
- Frequency of inspection, testing, or sampling plan
- Results of inspections and tests.

Inspection and test results are evaluated and verified by authorized personnel to document that all requirements have been satisfied. Records will identify the item examined, date of examination, examiner or data recorder, results, observations, acceptability (ensuring that all prerequisites for the given test are met), and action taken concerning any deviations noted.

Measuring and testing equipment used for inspections and tests is calibrated, maintained, and verified to be of the required precision and accuracy for use by personnel performing inspections and tests. Where possible, instruments have calibration certifications traceable to nationally recognized standards. Instruments are calibrated at specified intervals, before and after use, or just prior to use, as determined by required accuracy, intended use, frequency of use, stability characteristics, and other conditions affecting performance. Instruments are labeled, tagged, or otherwise controlled to indicate calibration status and to ensure traceability to calibration test data. Instruments found out-of-calibration or out-of-tolerance are tagged or segregated and not used until they are successfully recalibrated. The acceptability of items or processes measured, inspected, or tested with out-of-tolerance instruments is evaluated and measurements and tests repeated as required.

The SNF Project inspection and acceptance testing process meets the requirements of Part 2, Section 8, "Inspection and Acceptance Testing," of HNF-MP-599. Implementing procedures for inspection and acceptance testing are presented in the QAPP.

13.6.5 Assessments

This section briefly describes performance of management self-assessments and independent assessments to determine the adequacy of the quality program. The independent assessments are performed by Fluor Hanford, Incorporated (FH).

13.6.5.1 Management Assessments. Management assessments are planned, scheduled, and conducted by organization managers to regularly assess how well the organization is meeting its customers' requirements and expectations.

Management self-assessments also include the following activities:

- Processes including planning, organizational interfaces, use of performance indicators, training and qualifications, supervisory oversight, and support
- Routinely observing personnel performing activities and/or interviewing workers, reviewing documentation, and conducting drills or exercises
- Assessing performance indicators and other information for trends in improving or deteriorating conditions
- Measuring performance based on objective standards, clearly defined goals, review, and feedback processes.

Any barriers that are hindering the accomplishment of management objectives are identified, response actions documented, and corrective actions implemented.

The SNF Project management assessment process meets the requirements of Part 2, Section 9, "Management Assessments," of HNF-MP-599 and SNF Project administrative procedures. Implementing procedures for management assessments are presented in the QAPP.

13.6.5.2 Independent Assessments. The independent assessment program (1) provides for the measurement of item and service quality, requirements compliance, and work performance; and (2) promotes improvement. The assessment process incorporates a performance-based approach with emphasis on the results of work processes and compliance with requirements. Assessments are conducted on activities that most directly relate to final objectives and emphasize safety, reliability, and product performance.

The type of independent assessment performed and the frequency with which an assessment is performed are based on the status, complexity, and importance of the activity or process being assessed, and the past performance of the activity or process being assessed. Assessments shall be scheduled in a manner to provide coverage and coordination with ongoing quality assurance program activities. As a minimum, assessments should be performed on an annual basis. The assessment schedule shall be reviewed periodically and revised as necessary to ensure that coverage is maintained current. Independent assessments may include methods such as inspections, peer and technical reviews, audits, surveillance, and customer interviews. These independent assessments are conducted by technically qualified staff, knowledgeable in the activity or process being assessed, who have sufficient authority and freedom from the line organizations to carry out their responsibilities. An independent assessment evaluates the following areas:

- Work performance and process effectiveness
- Abnormal performance and potential problems
- Improvement opportunities

- Results documentation
- Satisfactory resolution of reported problems verification.

The independent assessment activity verifies by procedures, such as checking, assessing, and inspection, that activities affecting the functions important to safety have been correctly performed. Persons and organizations performing these quality assurance functions report to a management level to ensure that the required authority and organizational freedom are provided, including sufficient independence from cost and schedule considerations when these considerations are opposed to safety considerations.

Strengths and weaknesses identified by the independent assessment are documented and presented to the management of organizations responsible for performance of the subject activities or processes. Areas of weakness are then used to formulate corrective actions to promote improvements. Actions are tracked and the adequacy of corrective actions, including those taken to minimize or prevent recurrence, are verified. Lessons learned are communicated to other organizations with similar activities or concerns. Follow-up action, including a subsequent assessment of the activity or process, is initiated if appropriate to determine effectiveness of the corrective action.

The SNF Project independent assessment process meets the requirements of Part 2, Section 10, "Independent Assessments," of HNF-MP-599. Implementing procedures for independent assessments are presented in the QAPP.

13.6.6 Organizational Structure

FH is responsible to DOE, Richland Operations Office (DOE-RL), for planning, integrating, and managing SNF Project activities, including programs, projects, and operations. FH is supported by subcontractors, collectively referred to as the Project Hanford team. Chapter 17.0 of HNF-3553, *Spent Nuclear Fuel Project Final Safety Analysis Report* (HNF 1999), contains a detailed description of the SNF Project organization. In addition, other onsite contractors (e.g., Pacific Northwest National Laboratory, Hanford Site medical contractor), have existing contracts with DOE-RL for Hanford Site support services. Figure 13-1 depicts the Project Hanford organization; the roles of the DOE-RL, FH, Project Hanford Team, and enterprise companies are shown in Figure 13-2. As shown in Figures 13-1 and 13-2, the Project Hanford management and integration roles are divided between FH, major subcontractors, and the enterprise companies.

As noted in Figure 13-1, the FH Vice President, Environment, Safety, Health, and Quality Assurance, among others, reports directly to the president and chief executive officer of FH and provides oversight to ensure institutional safety provisions are implemented.

Figure 13-1. The Project Hanford Organization.

Fluor Hanford, Inc.

Figure 13-2. Project Hanford Management and Integration Roles.

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- Appendices as:
- Appendix A, "Multi-Canister Overpack to Multi-Canister Overpack Drop Analysis with Intermediate Impact Absorber"
 - Appendix B, "Cask Lid Accidental Drops onto Multi-Canister Overpack"
 - Appendix C, "Multi-Canister Overpack Drops into the Cask and onto the Cask Chamfer"
 - Appendix D, "Multi-Canister Overpack to Multi-Canister Overpack Drop Analysis"
 - Appendix E, "K-Basins Transportation Cask Operational and Accident Drop Analysis"
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 - Appendix H, "Simulation of SNF Transportation Cask Lid Drop onto MCO Process Port Cover Plate"
 - Appendix I, "Drop of Multi-Canister Overpack (MCO) into Transfer Cask"
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 - Appendix L, "Multi-Canister Overpack Drop Analyses and Engineering Check"
 - Appendix M, "Cask and Multi-Canister Overpack Drop onto the K-Basins Loadout Pit Curbing with Subsequent Horizontal Slap Down"
 - Appendix N, "K-Basin Immersion Pail Deceleration Capability for Cask Drop"
 - Appendix O, "Cask Drop Simulation into Canister Storage Building Transfer Pit"
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