FINAL SAFETY ANALYSIS REPORT

ON

THE HI-STAR 100 MPC STORAGE SYSTEM

By

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Certificate of Compliance (1008) and Final Safety Analysis Report Matrix

HI-STAR 100 Cask Storage System Final Safety Analysis Report (FSAR) Revision	NRC Certificate of Compliance (CoC) 1008 Amendment No.
0*	1
1	2
2**	2
3**	2
4	3
5**	3

* Revision 0 of this FSAR, issued in March 2001, included information supporting changes to CoC 72-1008 made in Amendment 1 (effective December 26, 2000), as well as information from the original version of the CoC that did not change as a result of that amendment. This is because the safety analysis report updating requirements of 10 CFR 72.248 did not become effective until after the original version of CoC 72-1008 became effective in October 1999. Therefore, a Final Safety Analysis Report (FSAR) was never issued to replace Revision 10 of the HI-STAR 100 Topical Safety Analysis Report (TSAR).

** These Revisions incorporated changes made via ECO/72.48 process only.

FSAR SECTION REVISION STATUS, LIST OF AFFECTED SECTIONS AND REVISION SUMMARY	
FSAR Report No.: HI-2012610	**FSAR Revision Number:** 5
FSAR Title:	Final Safety Analysis Report on the HI-STAR 100 System

This FSAR is submitted to the USNRC in support of Holtec International's application to secure a CoC under 10CFR Part 72.

FSAR review and verification are controlled at the chapter level and changes are annotated at the chapter level.

A section in a chapter is identified by two numerals separated by a decimal. Each section begins on a fresh page. Unless indicated as a "complete revision" in the summary description of change below, if any change in the content is made, then the change is indicated by a "bar" in the right page margin and the revision number of the entire chapter including applicable figures (annotated in the footer) is changed.

A summary description of change is provided below for each FSAR chapter. Minor editorial changes to this FSAR may not be summarized in the description of change.

Chapter 1

Affected Section or Table No.	Current Revision No.	Summary Description of Change
Section 1.5	Rev 5	Updated to identify applicable ECOs / 72.48s

Chapter 2

Section or Table No.	Current Revision No.	Summary Description of Change
N/A	Rev 4	

Chapter 3

Section or Table No.	Current Revision No.	Summary Description of Change
N/A	Rev 4	

Chapter 4		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
N/A	Rev 4	

Chapter 5		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
N/A	Rev 4	

Chapter 6		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
N/A	Rev 4	

Chapter 7		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
N/A	Rev 4	

Chapter 8		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
N/A	Rev 4	

Chapter 9		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
N/A	Rev 4	

Chapter 10		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
N/A	Rev 4	

Chapter 11		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
N/A	Rev 4	

Chapter 12		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
N/A	Rev 4	

Chapter 13		
Section or Table No.	**Current Revision No.**	**Summary Description of Change**
N/A	Rev 4	

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CHAPTER 1: GENERAL DESCRIPTION

1.0 GENERAL INFORMATION

This Final Safety Analysis Report (FSAR) for Holtec International's HI-STAR 100 System is a compilation of information and analyses to support a United States Nuclear Regulatory Commission (NRC) licensing review as a spent nuclear fuel dry storage cask under the requirements specified in 10CFR72 [1.0.1]. This FSAR describes the basis for NRC approval and issuance of a Certificate of Compliance (C of C) for storage under provisions of 10CFR72, Subpart L for the HI-STAR 100 to safely store spent nuclear fuel (SNF) at an Independent Spent Fuel Storage Installation (ISFSI) facility. This report has been prepared in the format and content suggested in NRC Regulatory Guide 3.61 [1.0.2] and NUREG-1536 Standard Review Plan for Dry Cask Storage Systems [1.0.3] to facilitate the NRC review process. To avoid restructuring of this report, Revision 4 does not follow the format presented in Revision 1 of NUREG-1536; however, the major technical updates suggested in Revision 1 of NUREG-1536 have been incorporated in this report.

The purpose of this chapter is to provide a general description of the design features and storage capabilities of the HI-STAR 100 System, drawings of the structures, systems, and components important to safety, and the qualifications of the certificate holder. This report is also suitable for incorporation into a site-specific Safety Analysis Report which may be submitted by an applicant for a site-specific 10 CFR 72 license to store SNF at an ISFSI or a facility similar in objective and scope. Table 1.0.1 contains a listing of the terminology and notation used in this FSAR.

To aid NRC review, additional tables and references have been added to facilitate the location of information requested by NUREG-1536. Table 1.0.2 provides a matrix of the topics in NUREG-1536 and Regulatory Guide 3.61, the corresponding 10CFR72 requirements, and a reference to the applicable FSAR section that addresses each topic.

The generic design basis and the corresponding safety analysis of the HI-STAR 100 System contained in this FSAR are intended to bound the SNF characteristics, design, conditions, and interfaces that exist in the vast majority of domestic power reactor sites and potential away-from-reactor storage sites in the contiguous United States. This FSAR also provides the basis for component fabrication and acceptance, and the requirements for safe operation and maintenance of the components, consistent with the design basis and safety analysis documented herein. In accordance with 10CFR72, Subpart K, site-specific implementation of the generically certified HI-STAR 100 System requires that the licensee perform a site-specific evaluation, as defined in 10CFR72.212. The HI-STAR 100 System FSAR identifies a limited number of conditions that are necessarily site-specific and are to be addressed in the licensee’s 10CFR72.212 evaluation. These include:

- Siting of the ISFSI and design of the storage pad and security system. Site-specific demonstration of compliance with regulatory dose limits. Implementation of a site-specific ALARA program.

- An evaluation of site-specific hazards and design conditions that may exist at the ISFSI site or the transfer route between the plant's cask receiving bay and the ISFSI. These include, but are not limited to, explosion and fire hazards, flooding conditions, land slides, and lightning protection.

- Determination that the physical and nucleonic characteristics and the condition of the SNF assemblies to be dry stored meet the fuel acceptance requirements of the Certificate of Compliance.

- An evaluation of interface and design conditions that exist within the plant's fuel building in which canister fuel loading, canister closure, and cask handling operations are to be conducted in accordance with the applicable 10CFR50 requirements and technical specifications for the plant.

- Detailed site-specific operating, maintenance, and inspection procedures prepared in accordance with the generic procedures and requirements provided in Chapters 8 and 9, and the technical specifications provided in the Certificate of Compliance.

- Performance of pre-operational testing.

- Implementation of a safeguards and accountability program in accordance with 10CFR73. Preparation of a physical security plan in accordance with 10CFR73.55.

- Review of the reactor emergency plan, quality assurance (QA) program, training program, and radiation protection program.

The generic safety analyses contained in the HI-STAR 100 FSAR may be used as input and for guidance by the licensee in performing a 10CFR72.212 evaluation.

Within this report, all figures, tables and references cited are identified by the double decimal system m.n.i, where m is the chapter number, n is the section number, and i is the sequential number. Thus, for example, Figure 1.2.3 is the third figure in Section 1.2 of Chapter 1. Wherever multiple units are shown, the US units are the governing value, and the SI units are shown for information only.

Revision 0 of this FSAR, issued in March 2001, included information supporting changes to CoC 72-1008 made in Amendment 1 (effective December 26, 2000), as well as information from the original version of the CoC that did not change as a result of that amendment. This is because the safety analysis report updating requirements of 10 CFR 72.248 did not become effective until after the original version of CoC 72-1008 became effective in October 1999. Therefore, a Final Safety Analysis Report (FSAR) was never issued to replace Revision 10 of the HI-STAR 100 Topical Safety Analysis Report (TSAR).

The following updates have been made to this FSAR in Revision 4 to align it with the HI-STAR 100 transport docket, the HI-STORM 100 which utilizes the same MPC models, and with HI-STORM FW and HI-STORM UMAX FSARs where the latest NRC regulatory positions are best articulated. Table 1.0.3 lists the NRC SERs in which the update has been approved in previous HI-STORM or HI-STAR dockets. In particular:

a. The list of MPCs admissible in HI-STAR 100 storage overpack (Table 1.1.1) has been aligned with that previously licensed for the same cask in the transport docket and the HI-STORM 100 storage docket. Specifically, MPC-32 for storing PWR fuel has been included.

b. The structural requirements to designate a cask transporter as single failure-proof have been excerpted from the HI-STORM FSARs to ensure consistency in the design criteria for the transporter which is typically used to handle both HI-STORMs and HI-STARs.

c. Standard system features and ancillaries such as the Forced Helium Dehydrator and the Metamic neutron absorber, which have been widely used in the HI-STORM dockets, are incorporated by reference in this docket for completeness.

d. Horizontal storage of the HI-STAR 100 casks is included. The safety analysis of the horizontal configuration relies on the analyses approved for its horizontal transport in the transportation docket.

e. The confinement boundary criteria are revised in accordance with ISG-18 to align with the criteria for the identical canisters in HI-STORM 100.

f. Soluble boron credit is taken for MPC-24 and MPC-32 in criticality analyses to align with the identical canisters in HI-STORM 100.

g. The quality assurance program information has been updated in accordance with the most up to date information in other Holtec dockets.

h. Pocket trunnions have been made optional, as the HI-STAR 100 can be rotated using a custom designed "rotator cradle," which entails a lower operational dose. The pocket trunnions are already optional under the HI-STAR 100 transportation license.

i. The permissible fuel cladding temperature limits for moderate and high burn up fuel under normal, accident and short term operating conditions have been updated to meet ISG-11 Rev 3 [Reference 1.0.6].

j. Miscellaneous editorial changes have been made to align the HI-STAR 100 FSAR with other Holtec dockets.

As in all HI-STAR and HI-STORM dockets, the grouping of canisters for the overpack are qualified for each Design Basis Loading (DBL) by selecting the canister whose response would be most limiting. Adding MPC-32 to the group means that the safety analysis under each DBL must be revisited and the analysis re- benchmarked if MPC-32 is indicated to be governing from prior analyses. For this purpose, the results from the safety analysis in the HI-STORM 100 FSAR and HI-STAR 100 transport SAR (which have all of the same MPCs) are consulted. As discussed in the subsequent chapters, the prior safety analyses of MPC-32 and their NRC approvals in HI-STAR transport and HI-STORM dockets provide a robust basis for incorporating it in the HI-STAR 100 storage docket by direct reference.

Thus, for example, MPC-32 is already certified for the design basis heat load specified herein under the transport docket with a complete set of qualifying calculations. Therefore, there is no need to upgrade the analysis to invoke modern CFD models on FLUENT as has been done in other dockets to increase the design basis heat load. Additionally, the criticality safety of MPC-32 has been previously established in the HI-STORM 100 docket, so those analyses are included herein, as described in Chapter 6. Material considerations remain unaffected because both HI-STAR 100 and the MPCs have been previously evaluated for material selection in the previously approved FSARs. Finally, the radiation safety, manufacturing, accident conditions and operations chapters are only minimally affected by the changes proposed in this LAR.

Table 1.0.3 provides a summary of prior SERs approving elements of the MPCs that are directly applicable to this safety analysis and are accordingly invoked in this FSAR

Indeed, since MPC-32 has been evaluated for the entire range of safety criteria in the HI-STORM 100 (storage) and HI-STAR 100 (transport) dockets, the safety evaluations documented in this FSAR leverage prior NRC-reviewed analyses: aside from minor manual calculations to address miscellaneous conditions, no new analyses were required to be performed to reach the safety conclusions documented in this issue of this FSAR. Those safety analyses which are made by reference to prior FSARs on another docket are appropriately identified to establish a clear nexus between this FSAR and the referenced FSAR. Additionally, the NRC approvals of these previous analyses as documented in the safety evaluation reports (SERs) are cited to the extent possible to anchor the safety case on NRC's prior evaluations.

1.0.1 Engineering Change Orders

Revision 4 of the HI-STAR 100 FSAR contains the changes related to Amendment 3 approved by USNRC and General FSAR changes authorized by Holtec ECOs (with corresponding 10CFR72.48 evaluations, if applicable). The changes authorized by the Holtec ECOs made to drawings listed in Section 1.5 and the general FSAR text ECOs are reflected in this Revision of the FSAR and listed in the following table. Note, drawing changes made under the listed ECOs were still submitted to the NRC and reviewed and approved as part of Amendment 3.

LIST OF ECO'S AND APPLICABLE 10CFR72.48 EVALUATIONS

Affected Item	ECO Number	72.48 Evaluation or Screening Number
HI-STAR 100		
MPC Enclosure Vessel	1021-149, 1022-107, 1023-86 1023-88	1407 1422
General FSAR Changes	5014-338	N/A

Table 1.0.1

TERMINOLOGY AND NOTATION

ALARA is an acronym for As Low As Reasonably Achievable.

Boral^{TM} means a rolled composite plate of aluminum and Boron carbide.

BWR is an acronym for boiling water reactor.

C.G. is an acronym for center of gravity.

Confinement Boundary means the outline formed by the sealed, cylindrical enclosure of the multi-purpose canister (MPC) shell welded to a solid baseplate, a lid welded around the top circumference of the shell wall, the port cover plates welded to the lid, and the closure ring welded to the lid and MPC shell.

Confinement System means the HI-STAR 100 multi-purpose canister (MPC) which encloses and confines the spent nuclear fuel during storage.

Controlled Area means that area immediately surrounding an ISFSI for which the owner/user exercises authority over its use and within which operations are performed.

DBE means Design Basis Earthquake.

DCSS is an acronym for Dry Cask Storage System.

Damaged Fuel Assembly is defined as a fuel assembly with known or suspected cladding defects, as determined by a review of records, greater than pinhole leaks or hairline cracks, missing fuel rods that are not replaced with dummy fuel rods, or those that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

Damaged Fuel Container means a specially designed enclosure for damaged fuel or fuel debris which permits gaseous and liquid media to escape while minimizing dispersal of gross particulates. DFCs authorized for use in the HI-STAR 100 System are the Holtec design or the Transnuclear Dresden Unit 1 design.

Enclosure Vessel means the pressure vessel defined by the cylindrical shell, baseplate, port cover plates, lid, and closure ring which provides confinement for the helium gas contained within the MPC. The Enclosure Vessel (EV) and the fuel basket together constitute the multi-purpose canister.

Fuel Basket means a honeycomb structural weldment with square openings that can accept a fuel assembly of the type for which it is designed.

Fuel Debris is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

TERMINOLOGY AND NOTATION

FSAR is an acronym for Final Safety Analysis Report (10CFR72).

Helium Retention Boundary means the enclosure formed by the overpack inner shell welded to a bottom plate and top main flange plus the bolted closure plate and port plugs with metallic seals. The helium retention boundary is an additional independent confinement boundary, however, no credit is taken for this additional barrier. The helium retention boundary maintains an inert helium atmosphere around the MPC.

HI-STAR 100 MPC means the sealed spent nuclear fuel container which consists of a honeycombed fuel basket contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. MPC is an acronym for multi-purpose canister. There are different MPCs with different fuel basket geometries for storing PWR or BWR fuel, but all MPCs have identical exterior dimensions. The MPC is the confinement boundary for storage conditions.

HI-STAR 100 overpack or overpack means the cask that receives and contains the sealed multi-purpose canisters containing spent nuclear fuel. It provides the retention boundary for the helium atmosphere, gamma and neutron shielding, and a set each of lifting and optional pocket trunnions. It is not defined as the confinement boundary for the radioactive material during storage.

HI-STAR 100 System consists of the HI-STAR 100 MPC sealed within the HI-STAR 100 overpack.

Holtite™ is the trade name for all present and future neutron shielding materials formulated under Holtec International’s R&D program dedicated to developing shielding materials for application in dry storage and transport systems. The Holtite development program is an ongoing experimentation effort to identify neutron shielding materials with enhanced shielding and temperature tolerance characteristics. Holtite-A™ is the first, and only shielding material qualified under the Holtec R&D program. As such, the terms Holtite and Holtite-A may be used inter changeably throughout this FSAR.

Holtite-A™ is a trademarked Holtec International neutron shield material.

Important to Safety (ITS) means a function or condition required to store spent nuclear fuel safely; to prevent damage to spent nuclear fuel during handling and storage, and to provide reasonable assurance that spent nuclear fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

Independent Spent Fuel Storage Installation (ISFSI) means a facility designed, constructed, and licensed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage in accordance with 10CFR72.

Intact Fuel Assembly is defined as a fuel assembly without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. Partial fuel assemblies, that is fuel assemblies from which fuel rods are missing, shall not be classified as intact fuel assemblies unless dummy fuel rods used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s).

Maximum Reactivity means the highest possible k-effective including bias, uncertainties, and calculational statistics evaluated for the worst-case combination of fuel basket manufacturing tolerances.

Metamic® *is a trade name for an aluminum/boron carbide composite neutron absorber material qualified for use in the MPCs.*

MGDS is an acronym for Mined Geological Depository System.

Multi-Purpose Canister (MPC) means the sealed canister which consists of a honeycombed fuel basket for spent nuclear fuel storage, contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. There are different MPCs with different fuel basket geometries for storing PWR or BWR fuel, but all MPCs have identical exterior dimensions. The MPC is the confinement boundary for storage conditions. MPC is an acronym for multi-purpose canister. The MPCs used as part of the HI-STORM 100 System (Docket No. 72-1014) are identical to the HI-STAR 100 MPCs evaluated in the HI-STAR 100 storage (Docket No. 72-1008) and transport (Docket No. 71-9261) applications.

MPC Fuel Basket means the honeycombed composite cell structure utilized to maintain subcriticality of the spent nuclear fuel. The number and size of the storage cells depends on the type of spent nuclear fuel to be stored. Each MPC fuel basket has sheathing welded to the storage cell walls for retaining the neutron absorber.

Neutron Shielding means Holtite or Holtite-A, a material used in the HI-STAR overpack to thermalize and capture neutrons emanating from the radioactive spent nuclear fuel.

PWR is an acronym for pressurized water reactor.

Reactivity is used synonymously with effective multiplication factor or k-effective.

SAR is an acronym for Safety Analysis Report (10CFR71).

Single Failure Proof means that the handling system is designed so that a single failure will not result in the loss of the capability of the system to safely retain the load.

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

SNF is an acronym for spent nuclear fuel.

SSC is an acronym for Structures, Systems and Components.

STP is Standard Temperature (298ºK) and Pressure (1 atm) conditions.

ZPA is an acronym for zero period acceleration.

Table 1.0.2

HI-STAR 100 SYSTEM FSAR REGULATORY COMPLIANCE CROSS-REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria(6)	Applicable 10CFR72 or 10CFR20 Requirement	HI-STAR FSAR
1. General Description			
1.1 Introduction	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.1
1.2 General Description	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.2
1.2.1 Cask Characteristics	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.2.1
1.2.2 Operational Features	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.2.2
1.2.3 Cask Contents	1.III.3 DCSS Contents	10CFR72.2(a)(1) 10CFR72.236(a)	1.2.3
1.3 Identification of Agents & Contractors	1.III.4 Qualification of the Applicant	10CFR72.24(j) 10CFR72.28(a)	1.3
1.4 Generic Cask Arrays	1.III.1 General Description & Operational Features	10CFR72.24(c)(3)	1.4
1.5 Supplemental Data	1.III.2 Drawings	10CFR72.24(c)(3)	1.5
NA	1.III.6 Consideration of Transport Requirements	10CFR72.230(b) 10CFR72.236(m)	1.1
NA	1.III.5 Quality Assurance	10CFR72.24(n)	1.3
2. Principal Design Criteria			
2.1 Spent Fuel To Be Stored	2.III.2.a Spent Fuel Specifications	10CFR72.2(a)(1) 10CFR72.236(a)	2.1
2.2 Design Criteria for Environmental Conditions and Natural Phenomena	2.III.2.b External Conditions, 2.III.3.b Structural, 2.III.3.c Thermal	0CFR71.122(b)	2.2
		10CFR72.122(c)	2.2.3.3, 2.2.3.10
		10CFR72.122(b)(1)	2.2
		10CFR72.122(b)(2)	2.2.3.11
		10CFR72.122(h)(1)	2.0
2.2.1 Tornado and Wind Loading	2.III.2.b External Conditions	10CFR72.122(b)	2.2.3.5
2.2.2 Water Level (Flood)	2.III.2.b External Conditions	10CFR72.122(b)(2)	2.2.3.6

HI-STAR 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS-REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria(6)	Applicable 10CFR72 or 10CFR20 Requirement	HI-STAR FSAR
	2.III.3.b Structural		
2.2.3 Seismic	2.III.3.b Structural	10CFR72.102(f) 10CFR72.122(b)(2)	2.2.3.7
2.2.4 Snow and Ice	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b)	2.2.1.6
2.2.5 Combined Load	2.III.3.b Structural	10CFR72.24(d) 10CFR72.122(b)(2)(ii)	2.2.7
NA	2.III.1 Structures, Systems, and Components Important to Safety	10CFR72.122(a) 10CFR72.24(c)(3)	2.2.4
NA	2.III.2 Design Criteria for Safety Protection Systems	10CFR72.236(g) 10CFR72.24(c)(1) 10CFR72.24(c)(2) 10CFR72.24(c)(4) 10CFR72.120(a) 10CFR72.236(b)	2.0, 2.2
NA	2.III.3.c Thermal	10CFR72.128(a)(4)	2.3.2.2, 4.0
NA	2.III.3.f Operating Procedures	10CFR72.24(f) 10CFR72.128(a)(5)	10.0, 8.0
		10CFR72.236(h)	8.0
		10CFR72.24 (l)(2)	1.2.1, 1.2.2
		10CFR72.236(l)	2.3.2.1
		10CFR72.24(e) 10CFR72.104(b)	10.0, 8.0
	2.III.3.g Acceptance Tests & Maintenance	10CFR72.122 (l) 10CFR72.236 (g) 10CFR72.122 (f) 10CFR72.128 (a)(1)	9.0
2.3 Safety Protection Systems	--	--	2.3
2.3.1 General	--	--	2.3
2.3.2 Protection by Multiple Confinement Barriers and Systems	2.III.3.b Structural	10CFR72.236(1)	2.3.2.1
	2.III.3.c Thermal	10CFR72.236(f)	2.3.2.2
	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.126(a) 10CFR72.128(a)(2)	2.3.5.2
		10CFR72.128(a)(3)	2.3.2.1
		10CFR72.236(d)	2.3.2.1, 2.3.5.2
		10CFR72.236(e)	2.3.2.1

Table 1.0.2 (continued)

HI-STAR 100 SYSTEM FSAR REGULATORY COMPLIANCE CROSS-REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria(6)	Applicable 10CFR72 or 10CFR20 Requirement	HI-STAR FSAR
2.3.3 Protection by Equipment & Instrument Selection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.122(h)(4) 10CFR72.122(i) 10CFR72.128(a)(1)	2.3.5
2.3.4 Nuclear Criticality Safety	2.III.3.e Criticality	10CFR72.124(a) 10CFR72.236(c) 10CFR72.124(b)	2.3.4, 6.0
2.3.5 Radiological Protection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	10.4.1
		10CFR72.24(d) 10CFR72.106(b) 10CFR72.236(d)	10.4.2
		10CFR72.24(m)	2.3.2.1
2.3.6 Fire and Explosion Protection	2.III.3.b Structural	10CFR72.122(c)	2.3.6, 2.2.3.10
2.4 Decommissioning Considerations	2.III.3.h Decommissioning	10CFR72.24(f) 10CFR72.130 10CFR72.236 (h)	2.4
	14.III.1 Design	10CFR72.130	2.4
	14.III.2 Cask Decontamination	10CFR72.236(i)	2.4
	14.III.3 Financial Assurance & Record Keeping	10CFR72.30	(1)
	14.III.4 License Termination	10CFR72.54	(1)
3. Structural Evaluation			
3.1 Structural Design	3.III.1 SSC Important to Safety	10CFR72.24(c)(3) 10CFR72.24(c)(4)	3.1
	3.III.6 Concrete Structures	10CFR72.182 (b) 10CFR72.182 (c)	3.1
3.2 Weights and Centers of Gravity	3.V.1.b.2 Structural Design Features	--	3.2
3.3 Mechanical Properties of Materials	3.V.1.c Structural Materials 3.V.2.c Structural Materials	10CFR72.24(c)(3)	3.3
NA	3.III.2 Radiation Shielding, Confinement, and Subcriticality	10CFR72.24(d) 10CFR72.124(a) 10CFR72.236(c) 10CFR72.236(d) 10CFR72.236(l)	3.4.4.3 3.4.7.3 3.4.10
NA	3.III.3 Ready Retrieval	10CFR72.122(f) 10CFR72.122(h)	3.4.4.3

HI-STAR 100 SYSTEM FSAR REGULATORY COMPLIANCE CROSS-REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria(6)	Applicable 10CFR72 or 10CFR20 Requirement	HI-STAR FSAR
		10CFR72.122(l)	
NA	3.III.4 Design-Basis Earthquake	10CFR72.24(c) 10CFR72.236(g)	3.4.7
NA	3.III.5 20 Year Minimum Design Length	10CFR72.24(c) 10CFR72.182(b) 10CFR72.182(c)	3.4.11 3.4.12
3.4 General Standards for Casks	--	--	3.4
3.4.1 Chemical and Galvanic Reactions	3.V.1.b.2 Structural Design Features	--	3.4.1
3.4.2 Positive Closure	--	--	3.4.2
3.4.3 Lifting Devices	3.V.1.ii(4)(a) Trunnions	--	3.4.3
3.4.4 Heat	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b) 10CFR72.236(g)	3.4.4
3.4.5 Cold	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.102(f) 10CFR72.122(b) 10CFR72.122(c) 10CFR72.236(g)	3.4.5
3.5 Fuel Rods	--	10CFR72.122(h)(1)	3.5
4. Thermal Evaluation			
4.1 Discussion	4.III Regulatory Requirements	10CFR72.24(c)(3) 10CFR72.128(a)(4) 10CFR72.236(f) 10CFR72.236(h)	4.1, 4.5
4.2 Summary of Thermal Properties of Materials	4.V.4.b Material Properties	--	4.2
4.3 Specifications for Components	4.IV Acceptance Criteria	10CFR72.122(h)(1)	4.3
4.4 Thermal Evaluation for Normal Conditions of Storage	4.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.236(g)	4.4
NA	4.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.122(c)	11.1, 11.2
4.5 Supplemental Data	4.V.6 Supplemental Info.	--	--
5. Shielding Evaluation			
5.1 Discussion and Results	--	10CFR72.104(a) 10CFR72.106(b)	5.1
5.2 Source Specification	5.V.2 Radiation Source Definition	--	5.2

Table 1.0.2 (continued)

HI-STAR 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS-REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria(6)	Applicable 10CFR72 or 10CFR20 Requirement	HI-STAR FSAR
5.2.1 Gamma Source	5.V.2.a Gamma Source	--	5.2.1, 5.2.3
5.2.2 Neutron Source	5.V.2.b Neutron Source	--	5.2.2, 5.2.3
5.3 Model Specification	5.V.3 Shielding Model Specification	--	5.3, 5.3.3
5.3.1 Description of the Radial and Axial Shielding Configuration	5.V.3.a Configuration of the Shielding and Source	10CFR72.24(c)(3)	5.3.1
5.3.2 Shield Regional Densities	5.V.3.b Material Properties	10CFR72.24(c)(3)	5.3.2
5.4 Shielding Evaluation	5.V.4 Shielding Analysis	10CFR72.24(d) 10CFR72.104(a) 10CFR72.106(b) 10CFR72.128(a)(2) 10CFR72.236(d)	5.4
5.5 Supplemental Data	5.V.5 Supplemental Info.	--	Appendices 5.A, 5.B, and 5.C
6. Criticality Evaluation			
6.1 Discussion and Results	--	--	6.1
6.2 Spent Fuel Loading	6.V.2 Fuel Specification	--	6.1, 6.2
6.3 Model Specifications	6.V.3 Model Specification	--	6.3
6.3.1 Description of Calculational Model	6.V.3.a Configuration	10CFR72.124(b) 10CFR72.24(c)(3)	6.3.1
6.3.2 Cask Regional Densities	6.V.3.b Material Properties	10CFR72.24(c)(3) 10CFR72.124(b) 10CFR72.236(g)	6.3.2
6.4 Criticality Calculations	6.V.4 Criticality Analysis	10CFR72.124	6.4
6.4.1 Calculational or Experimental Method	6.V.4.a Computer Programs and 6.V.4.b Multiplication Factor	10CFR72.124	6.4.1
6.4.2 Fuel Loading or Other Contents Loading Optimization	6.V.3.a Configuration	--	6.4.2
6.4.3 Criticality Results	6.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.124 10CFR72.236(c)	6.1, 6.2, 6.3.1, 6.3.2
6.5 Critical Benchmark Experiments	6.V.4.c Benchmark Comparisons	--	6.5, Appendix 6.A, 6.4.3

HI-STAR 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS-REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria(6)	Applicable 10CFR72 or 10CFR20 Requirement	HI-STAR FSAR
6.6 Supplemental Data	6.V.5 Supplemental Info.		Appendices 6.B, 6.C, and 6.D
7. Confinement			
7.1 Confinement Boundary	7.III.1 Description of Structures, Systems, and Components Important to Safety	10CFR72.24(c)(3) 10CFR72.24(l)	7.0, 7.1
7.1.1 Confinement Vessel	7.III.2 Protection of Spent Fuel Cladding	10CFR72.122(h)(l)	7.1, 7.1.1, 7.2.2
7.1.2 Confinement Penetrations	--	--	7.1.2
7.1.3 Seals and Welds	--	--	7.1.3
7.1.4 Closure	7.III.3 Redundant Sealing	10CFR72.236(e)	7.1.1, 7.1.4
7.2 Requirements for Normal Conditions of Storage	7.III.7 Evaluation of Confinement System	10CFR72.24(d) 10CFR72.236(l)	7.2
7.2.1 Release of Radioactive Material	7.III.6 Release of Nuclides to the Environment	10CFR72.24(l)(1)	7.2.1
	7.III.4 Monitoring of Confinement System	10CFR72.122(h)(4) 10CFR72.128(a)(l)	7.1.4
	7.III.5 Instrumentation	10CFR72.24(l) 10CFR72.122(i)	7.1.4
	7.III.8 Annual Dose	10CFR72.104(a)	7.3.5
7.2.2 Pressurization of Confinement Vessel	--	--	7.2.2
7.3 Confinement Requirements for Hypothetical Accident Conditions	7.III.7 Evaluation of Confinement System	10CFR72.24(d) 10CFR72.122(b) 10CFR72.236(l)	7.3
7.3.1 Fission Gas Products	--	--	7.3.1
7.3.2 Release of Contents	--	--	7.3.3
NA	--	10CFR72.106(b)	7.3
7.4 Supplemental Data	7.V Supplemental Info.	--	--
8. Operating Procedures			
8.1 Procedures for Loading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	8.1 to 8.5
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	8.1.5
	8.III.3 Radioactive Effluent Control	10CFR72.24(l)(2)	8.1.5, 8.5.2

Table 1.0.2 (continued)

HI-STAR 100 SYSTEM FSAR REGULATORY COMPLIANCE CROSS-REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria[6]	Applicable 10CFR72 or 10CFR20 Requirement	HI-STAR FSAR
	8.III.4 Written Procedures	10CFR72.212(b)(9)	8.0
	8.III.5 Establish Written Procedures and Tests	10CFR72.234(f)	8.0
	8.III.6 Wet or Dry Loading and Unloading Compatibility	10CFR72.236(h)	8.0
	8.III.7 Cask Design to Facilitate Decon	10CFR72.236(i)	8.1, 8.3
8.2 Procedures for Unloading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	8.3
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	--
	8.III.3 Radioactive Effluent Control	10CFR72.24(l)(2)	8.3.3
	8.III.4 Written Procedures	10CFR72.212(b)(9)	8.0
	8.III.5 Establish Written Procedures and Tests	10CFR72.234(f)	8.0
	8.III.6 Wet or Dry Loading and Unloading Compatibility	10CFR72.236(h)	8.0
	8.III.8 Ready Retrieval	10CFR72.122(l)	8.3
8.3 Preparation of the Cask	--	--	8.3.2
8.4 Supplemental Data	--	--	Tables 8.1.1 to 8.1.10
NA	8.III.9 Design To Minimize Radwaste	10CFR72.24(f) 10CFR72.128(a)(5)	8.1, 8.3
	8.III.10 SSCs Permit Inspection, Maintenance, and Testing	10CFR72.122(f)	Table 8.1.6
9. Acceptance Criteria and Maintenance Program			
9.1 Acceptance Criteria	9.III.1.a Preoperational Testing & Initial Operations	10CFR72.24(p)	8.1, 9.1
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.24(c) 10CFR72.122(a)	9.1
	9.III.1.d Test Program	10CFR72.162	9.1

Table 1.0.2 (continued)

HI-STAR 100 SYSTEM FSAR REGULATORY COMPLIANCE CROSS-REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	**Associated NUREG-1536 Review Criteria[6]**	**Applicable 10CFR72 or 10CFR20 Requirement**	**HI-STAR FSAR**
	9.III.1.e Appropriate Tests	10CFR72.236(l)	9.1
	9.III.1.f Inspection for Cracks, Pinholes, Voids and Defects	10CFR72.236(j)	9.1
	9.III.1.g Provisions that Permit Commission Tests	10CFR72.232(b)	9.1[2]
9.2 Maintenance Program	9.III.1.b Maintenance	10CFR72.236(g)	9.2
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.122(f) 10CFR72.128(a)(l)	9.2
	9.III.1.h Records of Maintenance	10CFR72.212(b)(8)	9.2
NA	9.III.2 Resolution of Issues Concerning Adequacy of Reliability	10CFR72.24(i)	[3]
	9.III.1.d Submit Pre-Op Test Results to NRC	10CFR72.82(e)	[4]
	9.III.1.i Casks Conspicuously and Durably Marked	10CFR72.236(k)	9.1.7, 9.1.1.(12)
	9.III.3 Cask Identification		
10. Radiation Protection			
10.1 Ensuring that Occupational Exposures Are As Low As Reasonably Achievable (ALARA)	10.III.4 ALARA	10CFR20.1101 10CFR72.24(e) 10CCR72.104(b) 10CFR72.126(a)	10.1
10.2 Radiation Protection Design Features	10.V.1.b Design Features	10CFR72.126(a)(6)	10.2
10.3 Estimated Onsite Collective Dose Assessment	10.III.2 Occupational Exposures	10CFR20.1201 10CFR20.1207 10CFR20.1208 10CFR20.1301	10.3
NA	10.III.3 Public Exposure	10CFR72.104 10CFR72.106	
	10.III.1 Effluents and Direct Radiation	10CFR72.104	10.4
11. Accident Analyses			
11.1 Off-Normal Operations	11.III.2 Meet Dose Limits for Anticipated Events	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	11.1

HI-STAR 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS-REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria(6)	Applicable 10CFR72 or 10CFR20 Requirement	HI-STAR FSAR
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	11.1
	11.III.7 Instrumentation and Control for Off-Normal Condition	10CFR72.122(i)	11.1
11.2 Accidents	11.III.1 SSCs Important to Safety Designed for Accidents	10CFR72.24(d)(2) 10CFR72.122(b)(2) 10CFR72.122(b)(3) 10CFR72.122(d) 10CFR72.122(g)	11.2
	11.III.5 Maintain Confinement for Accident	10CFR72.236(l)	11.2
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	11.2, 6.0
	11.III.3 Meet Dose Limits for Accidents	10CFR72.24(d)(2) 10CFR72.24(m) 10CFR72.106(b)	11.2, 5.1.2, 7.3
	11.III.6 Retrieval	10CFR72.122(l)	8.3
	11.III.7 Instrumentation and Control for Accident Conditions	10CFR72.122(i)	(5)
NA	11.III.8 Confinement Monitoring	10CFR72.122(h)(4)	7.1.4
12. Operating Controls and Limits			
12.1 Proposed Operating Controls and Limits	--	10CFR72.44(c)	12.0
	12.III.1.e Administrative Controls	10CFR72.44(c)(5)	12.0
12.2 Development of Operating Controls and Limits	12.III.1 General Requirement for Technical Specifications	10CFR72.24(g) 10CFR72.26 10CFR72.44(c) 10CFR72 Subpart F 10CFR72 Subpart F	12.0
12.2.1 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings	12.III.1.a Functional/ Operating Units, Monitoring Instruments and Limiting Controls	10CFR72.44(c)(l)	Appendix 12.A
12.2.2 Limiting Conditions for Operation	12.III.1.b Limiting Controls	10CFR72.44(c)(2)	Appendix 12.A
	12.III.2.a Type of Spent Fuel	10CFR72.236(a)	Appendix 12.A
	12.III.2.b Enrichment		

HI-STAR 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS-REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria(6)	Applicable 10CFR72 or 10CFR20 Requirement	HI-STAR FSAR
	12.III.2.c Burnup		
	12.III.2.d Minimum Acceptable Cooling Time		
	12.III.2.f Maximum Spent Fuel Loading Limit		
	12.III.2.g Weights and Dimensions		
	12.III.2.h Condition of Spent Fuel		
	12.III.2.e Maximum Heat Dissipation	10CFR72.236(a)	Appendix 12.A
	12.III.2.i Inerting Atmosphere Requirements	10CFR72.236(a)	Appendix 12.A
12.2.3 Surveillance Specifications	12.III.1.c Surveillance Requirements	10CFR72.44(c)(3)	Chapter 12
12.2.4 Design Features	12.III.1.d Design Features	10CFR72.44(c)(4)	Chapter 12
12.2.5 Suggested Format for Operating Controls and Limits	--	--	Appendix 12.A
NA	12.III.2 SCC Design Bases and Criteria	10CFR72.236(b)	2.0
NA	12.III.2 Criticality Control	10CFR72.236(c)	2.3.4, 6.0
NA	12.III.2 Shielding and Confinement	10CFR20 10CFR72.236(d)	2.3.5, 7.0, 5.0, 10.0
NA	12.III.2 Redundant Sealing	10CFR72.236(e)	7.1, 2.3.2
NA	12.III.2 Passive Heat Removal	10CFR72.236(f)	2.3.2.2, 4.0
NA	12.III.2 20 Year Storage and Maintenance	10CFR72.236(g)	1.2.1.5, 9.0, 3.4.10, 3.4.11
NA	12.III.2 Decontamination	10CFR72.236(i)	8.0, 10.1
NA	12.III.2 Wet or Dry Loading	10CFR72.236(h)	8.0
NA	12.III.2 Confinement Effectiveness	10CFR72.236(j)	9.0
NA	12.III.2 Evaluation for Confinement	10CFR72.236(l)	7.1, 7.2, 9.0
13. Quality Assurance			

Table 1.0.2 (continued)

HI-STAR 100 SYSTEM FSAR REGULATORY COMPLIANCE CROSS-REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria(6)	Applicable 10CFR72 or 10CFR20 Requirement	HI-STAR FSAR
13.1 Quality Assurance	13.III Regulatory Requirements	10CFR72.24 (m)	13.0
	13.IV Acceptance Criteria	10CFR72, Subpart G	

Table 1.0.2 (continued)

HI-STAR 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS-REFERENCE MATRIX

Notes:

(1) The stated requirement is the responsibility of the licensee (i.e., utility) as part of the ISFSI pad and is therefore not addressed in this application.

(2) It is assumed that approval of the FSAR by the NRC is the basis for the Commission's acceptance of the tests defined in Chapter 9.

(3) Not applicable to HI-STAR 100 System. The functional adequacy of all-important to safety components is demonstrated by analyses.

(4) The stated requirement is the responsibility of licensee (i.e., utility) as part of the ISFSI and is therefore not addressed in this application.

(5) The stated requirement is not applicable to the HI-STAR 100 System. No monitoring is required for accident conditions.

(6) Revision 4 of this FSAR does not follow the format presented in Revision 1 of NUREG-1536; however, the major technical updates suggested in Revision 1 of NUREG-1536 have been incorporated in this report.

"--" There is no corresponding NUREG-1536 criteria, no applicable 10CFR72 or 10CFR20 regulatory requirement, or the item is not addressed in the FSAR.

"NA" There is no Regulatory Guide 3.61 section that corresponds to the NUREG-1536, 10CFR72, or 10CFR20 requirement being addressed.

Table 1.0.3: NRC SERs on Previous HI-STORM and HI-STAR Dockets			
Docket number & Cask	Item	SER #	Publication date
72-1014 (HI-STORM 100)	MPC-32 approved for storage of PWR fuel in HI-STORM 100	SER Amendment 1 [1.0.12]	July 15, 2002
71-9261 (HI-STAR 100 transport)	MPC-32 approved for transport of PWR fuel in HI-STAR 100	SER Amendment 5 [1.0.5]	Oct 12, 2006
72-1032 (HI-STORM FW)	Single failure proof handling devices approved	SER Amendment 0 [1.0.7]	July 14, 2011
71-9261 (HI-STAR 100 transport)	Use of Forced Helium Dehydration (FHD) approved	SER Amendment 2 [1.0.8]	May 31, 2002
71-9261 (HI-STAR 100 transport)	METAMIC neutron absorber (Use of METAMIC explicitly authorized)	SER Amendment 8 [1.0.9]	Oct 12, 2010
71-9261 (HI-STAR 100 transport)	Transport of the HI-STAR 100 in the horizontal position approved	SER Amendment 0 [1.0.13]	March 31, 1999
72-1014 (HI-STORM 100)	HI-STAR/HI-STORM MPCs ruled to be leak-tight in accordance with ISG-18	SER Amendment 2 [1.0.10]	June 7, 2005
72-1014 (HI-STORM 100)	Soluble boron credit for MPC-32 and MPC-24 approved	SER Amendment 1 [1.0.12]	July, 15, 2002
72-1040 (HI-STORM UMAX)	Most recent Quality Assurance Program description approved in UMAX docket.	SER Amendment 0 [1.0.11]	April 2, 2015
71-9261 (HI-STAR 100 transport)	Pocket trunnions made optional in HI-STAR 100 for transport	SER Amendment 2 [1.0.8]	May 31, 2002
All HI-STAR and HI-STORM dockets	Permissible fuel cladding temperature limits for moderate and high burn up fuel under normal, accident and short term operating conditions updated to meet ISG-11 Rev 3 [1.0.6].	N/A	N/A

1.1 INTRODUCTION

HI-STAR 100 (acronym for Holtec International Storage, Transport and Repository) is a spent nuclear fuel (SNF) storage and transport (dual purpose) system designed to be in general compliance with the U.S. Department of Energy's (DOE) design procurement specifications for multi-purpose canisters and large transportation casks [1.1.1], [1.1.2]. The annex "100" is a model number designation which denotes a system weighing in the range of 100 tons. The HI-STAR 100 System consists of a sealed metallic canister, herein abbreviated as the "MPC", contained within an overpack. Figure 1.1.1 depicts the HI-STAR 100.

The HI-STAR 100 System is designed to accommodate a wide variety of spent fuel assemblies in a single overpack by utilizing different MPCs. The external dimensions of all MPCs are identical to allow the use of a single overpack design. Each of the MPCs has different internals (baskets) to accommodate distinct fuel characteristics. Each MPC is identified by the maximum quantity of fuel assemblies it is capable of receiving. Thus, the MPC-24 can contain a maximum of 24 PWR assemblies and the MPC-68 can contain a maximum of 68 BWR assemblies. Figure 1.1.2 depicts the HI-STAR 100 with two of its major constituents, the MPC and the overpack, in a cutaway view.

As shown in Table 1.1.1, this revision of the FSAR incorporates MPC-32 to the list of canisters for the HI-STAR 100 used fuel storage system. MPC-32 can contain a maximum of 32 PWR assemblies. It is recalled that MPC-32 has been certified for transport in the HI-STAR 100 transport package in Docket #71-9261 since 2006.

Revision 4 of this FSAR also updates NRC's acceptance criteria that post- date the last issue of this FSAR such as ISG-11 Rev 3 for fuel cladding temperature limits.

The HI-STAR 100 is designed for both storage and transport. In the storage mode, HI-STAR can be positioned in either the vertical or horizontal or an inclined orientation. The safety analysis for HI-STAR 100 in the storage mode, presented in this FSAR accordingly either considers the most adverse orientation that will yield the minimum safety margin or evaluates both orientations. The HI-STAR 100 System's multi-purpose design reduces SNF handling operations and thereby enhances radiological protection. Once the SNF is loaded and the MPC and cask are sealed, the HI-STAR 100 System can be positioned on-site for temporary or long-term storage or transported directly off-site. The HI-STAR 100 System's ability to both store and transport SNF eliminates repackaging. It should be emphasized that, under transport conditions, the HI-STAR 100 cask provides an autonomous radiological containment boundary provided by the sealed overpack independent of the MPC as demonstrated in Reference [1.2.6]. Therefore, during storage, the concentric metallic seals in the overpack closure plate provide an additional barrier against release of radiological matter to the environment in HI-STAR 100 compared to the ventilated storage systems.

The HI-STAR 100 System is a completely passive stand-alone storage system which provides SNF confinement, radiation shielding, structural integrity, criticality control, and heat removal independent of any other facility, structures or components. This Final Safety Analysis Report (FSAR) provides bounding values for design criteria to facilitate NRC review and evaluation for both General License use under 10CFR72, Subpart K, and as reference for a site-specific storage facility application.

This FSAR demonstrates the inherent safety of one loaded overpack or multiple overpacks at an ISFSI. The HI-STAR 100 System can be used alone or as part of a multi-unit array at an ISFSI. The site for the ISFSI can be located either at a reactor or away from a reactor.

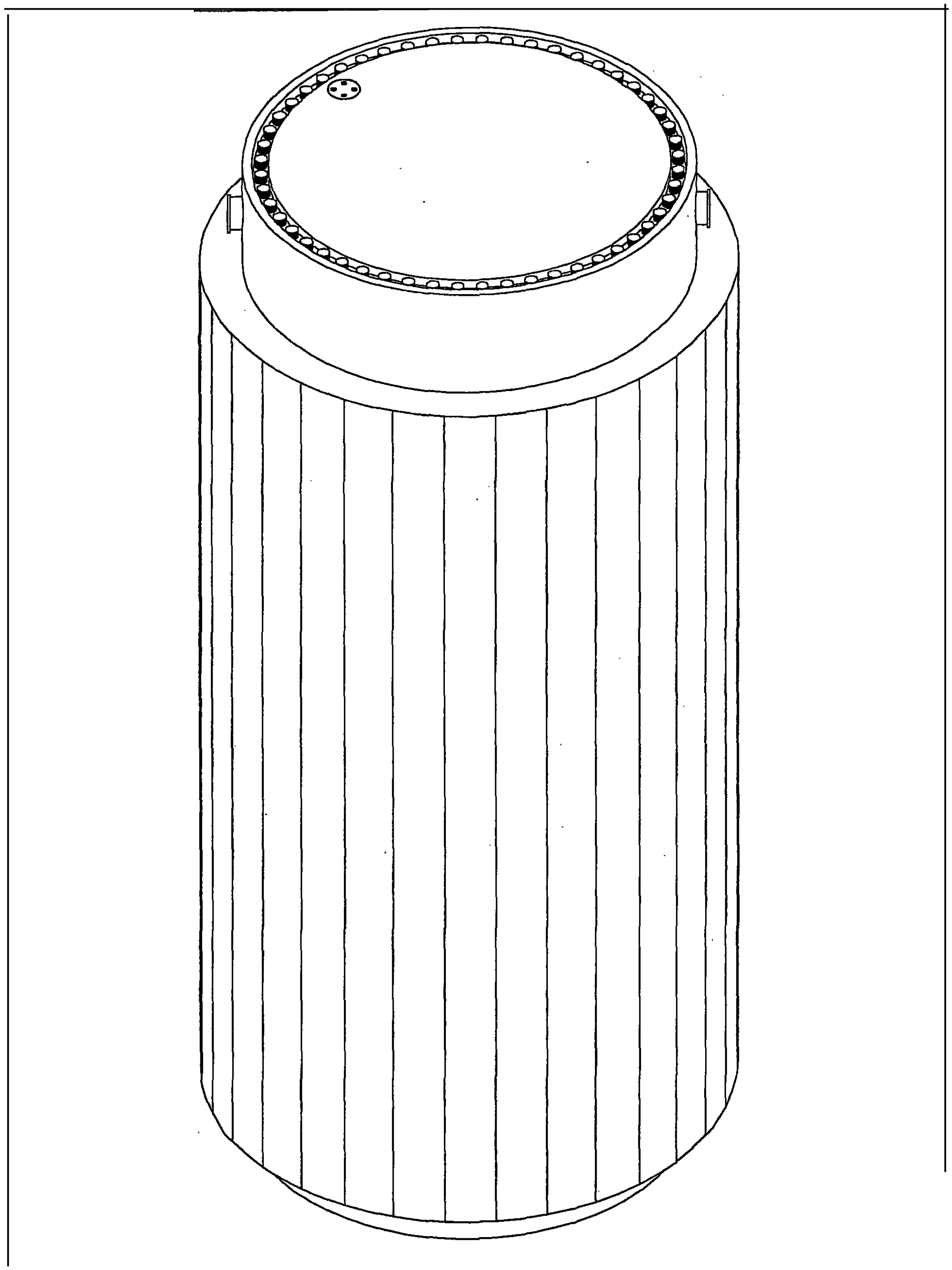


FIGURE 1.1.1; PICTORIAL VIEW OF HI-STAR 100

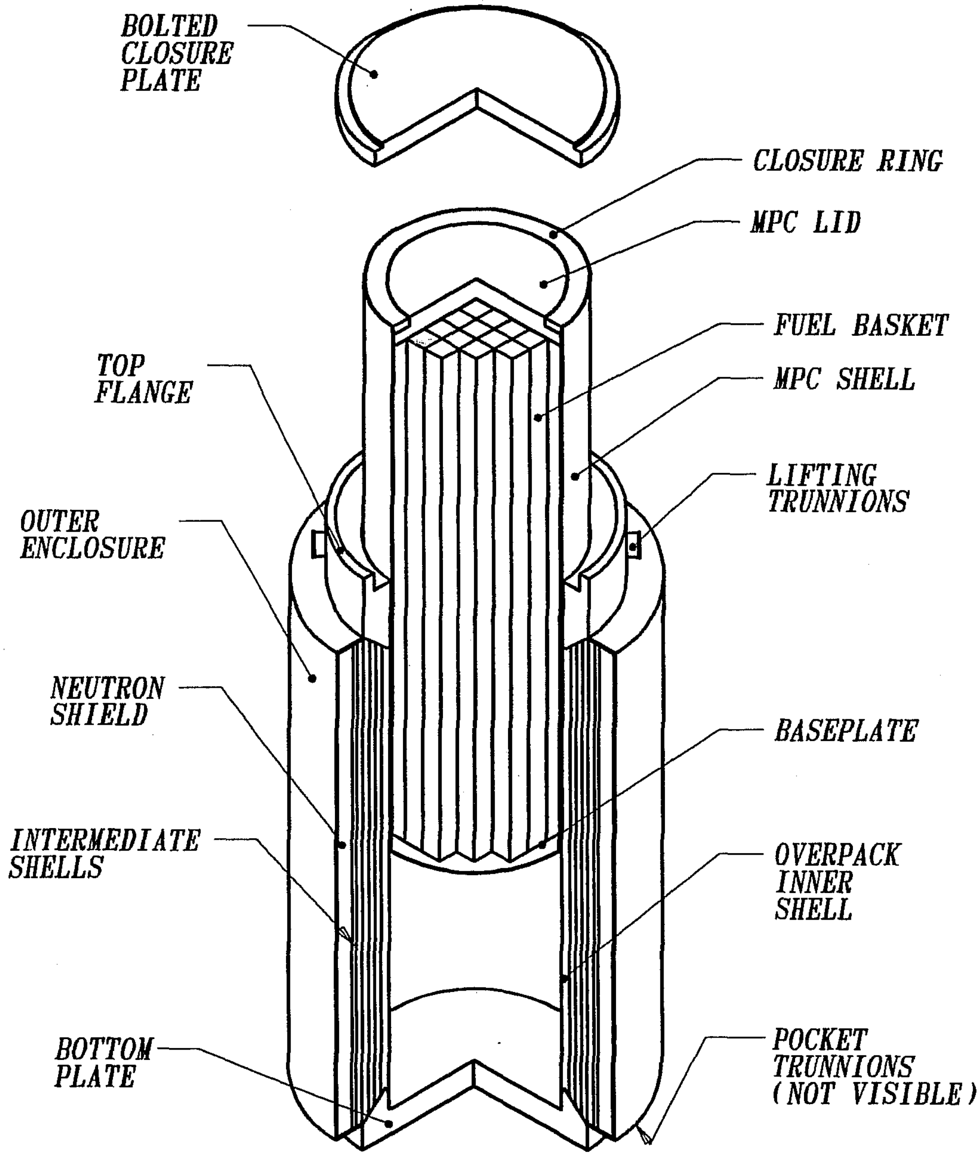


Figure 1.1.2; HI-STAR 100 OVERPACK WITH MPC PARTIALLY INSERTED

Table 1.1.1: MPC types and their Certification History			
MPC Name	Storage Capacity(cells)	Fuel Type	Comment
MPC-68/68F	68	BWR	Original certification
MPC-24	24	PWR	Original certification
MPC-32	32	PWR	Proposed for addition in the present FSAR

1.2 GENERAL DESCRIPTION AND OPERATING FEATURES OF HI-STAR 100

1.2.1 System Characteristics

The complete HI-STAR 100 System for storage of spent nuclear fuel is comprised of two discrete components:

- the multi-purpose canister (MPC), and
- the storage/transport overpack

Necessary auxiliaries required to deploy the HI-STAR 100 System for storage are:

- lifting and handling systems
- welding equipment
- vacuum drying system and helium backfill system with leak detector or a Forced Helium Dehydration system
- a heavy haul transfer device transporter
- support frame for horizontal storage orientation

The HI-STAR 100 System consists of interchangeable MPCs which constitute the confinement boundary for BWR or PWR spent nuclear fuel, and an overpack which provides the helium retention boundary. Tables 1.2.1 and 1.2.2 contain the key parameters for the HI-STAR 100 MPCs. Figure 1.2.1 provides a cross sectional elevation view of the HI-STAR 100 System in storage.

All MPCs have identical exterior dimensions which render them interchangeable. The dimensions of the MPC are given in the drawings listed in Section 1.5. Due to the differing storage contents of each of the MPCs, the maximum loaded weight differs between each MPC. However, the maximum weight of a loaded MPC is approximately 44-1/2 tons (40.4 metric tons).

A single overpack design is provided which is capable of storing each type of MPC. The dimensions of the overpack are given in the drawing listed in Section 1.5. The overpack inner cavity is sized to accommodate the MPCs. The weight of the overpack without an MPC is approximately 77 tons (70 metric tons).

Before proceeding to present detailed physical data on the HI-STAR 100 System, it is contextual to summarize the design attributes which set it apart from the prior generation of casks. There are several features in the HI-STAR 100 System design which increase its effectiveness with respect to the safe storage and transport of spent nuclear fuel (SNF). Some of the principal features of the HI-STAR 100 System which enhance its effectiveness as an SNF storage device and a safe SNF confinement structure are:

- the honeycomb design of the MPC fuel basket
- the effective distribution of neutron and gamma shielding materials within the system
- the high heat expulsion capability
- the structural robustness of the multi-shell overpack construction

The honeycomb design of the MPC fuel baskets renders the basket into a multi-flange plate weldment where all structural elements (box walls) are arrayed in two orthogonal sets of plates. Consequently, the walls of the cells are either completely co-planar (no offset) or orthogonal with each other. There is complete edge-to-edge continuity between the contiguous cells.

Among the many benefits of the honeycomb construction is the uniform distribution of the metal mass of the basket over the body of the basket (in contrast to the "box and spacer disk" construction where the support plates are localized mass points). Physical reasoning suggests that a uniformly distributed mass provides a more effective shielding barrier than can be obtained from a nonuniform (box and spacer disk) basket. In other words, the honeycomb basket is a more effective radiation attenuation device.

The complete cell-to-cell connectivity inherent in the honeycomb basket structure provides an uninterrupted heat transmission path, making the HI-STAR 100 MPC an effective heat rejection device.

Finally, the multilayer shell construction in the overpack provides a natural barrier against crack propagation in the radial direction through the overpack structure. If, during a mechanical accident (drop) event, a crack was initiated in one layer, the crack could not propagate to the adjacent layer. Additionally, it is less likely that a crack would initiate as the thinner layers are more ductile than a thicker plate.

A description of each of the HI-STAR components is provided in the following subsections, along with information with respect to its fabrication and safety features. This discussion is supplemented with the full set of drawings in Section 1.5.

1.2.1.1 Multi-Purpose Canisters

The HI-STAR 100 MPCs are welded cylindrical structures with flat ends as shown in cross sectional views of Figures 1.2.2 and 1.2.4. Each MPC is an assembly consisting of a honeycombed fuel basket, a baseplate, canister shell, a lid, and a closure ring, as depicted in the MPC cross section elevation view, Figure 1.2.5. The outer diameter and cylindrical height of each MPC is fixed. However, the number of spent nuclear fuel storage locations in each of the MPCs depends on the fuel assembly characteristics. Drawings of the MPCs are provided in Section 1.5.

The MPC provides the confinement boundary for the stored fuel. Figure 1.2.6 provides an elevation view of the MPC confinement boundary. The confinement boundary is a seal-welded enclosure constructed entirely of stainless steel.

The construction features of the PWR MPC-24, PWR MPC-32, and the BWR MPC-68 are similar. However, the PWR MPC-24 canister in Figure 1.2.4, which is designed for highly enriched PWR fuel without credit for soluble boron, differs in construction from the MPC-68 in one important aspect: The fuel storage cells are physically separated from one another by a "flux trap" between each storage cell for criticality control. All MPC baskets are formed from an array of plates welded to each other, such that a honeycomb structure is created which resembles a multiflanged, closed-section beam in its structural characteristics. MPC-32 contains a non- flux trap (NFT) fuel basket for

PWR fuel, shown in the Licensing drawing package in Section 1.5. Anatomically, MPC-32 fuel basket emulates MPC-68 (for BWR fuel) in all key respects. The similarities and differences are summarized below:

a. The basket is composed of a honeycomb grid work of Alloy X plates arranged on a square pitch. The plate thickness and the cell size in MPC-32 are larger than MPC-68. Correspondingly, the cell count is lower (32 vs 68).
b. A panel of a qualified neutron absorber material (such as Metamic) is installed on the walls of the cell such that there is one neutron absorber panel between two adjacent storage cavities.
c. A formed sheathing made of Alloy X envelopes each neutron absorber panel providing it locational fixity for criticality control.
d. Basket shims are installed in the space between the enclosure vessel and the periphery of the basket to undergird it laterally when the cask is in the horizontal configuration.
e. Axial spacers are utilized, as needed, to limit the axial movement of the fuel within the MPC.
f. The plates that comprise the fuel basket are welded to each other (longitudinal fillet welds) at their intersections to create a robust honeycomb structure.

The MPC fuel basket is positioned and supported within the MPC shell by "basket shims" located in the space between the inside of the MPC shell and the Fuel Basket. Between the periphery of the basket, the MPC shell, and the basket supports, heat conduction elements are optionally installed. These heat conduction elements are fabricated from thin aluminum alloy 1100 in shapes and a design which allow a snug fit in the confined spaces and ease of installation. The heat conduction elements are installed along the full length of the MPC basket, except at the drain pipe location, to create a nonstructural thermal connection which facilitates heat transfer from the basket to shell. In their operating condition, the heat conduction elements will conform to and contact the MPC shell and basket walls.

Lifting lugs attached to the inside surface of the MPC canister shell serve to permit lifting and placement of the empty MPC into the overpack. The lifting lugs also serve to axially locate the lid prior to welding. These internal lifting lugs are not used to handle a loaded MPC. Since the MPC lid is installed prior to any handling of the loaded MPC, there is no access to the lifting lugs once the MPC is loaded.

The top end of the HI-STAR 100 MPC incorporates a redundant closure system. Figure 1.2.6 provides a sketch of the MPC closure details. The MPC lid is a circular plate (fabricated from one piece, or two pieces - split top and bottom) edge-welded to the MPC outer shell. If the two-piece lid design is employed, only the top piece is analyzed as part of the enclosure vessel pressure boundary. The bottom piece acts as a radiation shield and is attached to the top piece with a non-structural, non-pressure retaining weld. This lid is equipped with vent and drain ports which are utilized to remove moisture and air from the MPC, and backfill the MPC with a specified pressure of inert gas (helium). The vent and drain ports are covered and welded before the closure ring is installed. The closure ring is a circular ring edge-welded to the MPC shell and lid. The MPC lid provides sufficient rigidity to allow the entire MPC loaded with SNF to be lifted by threaded holes in the MPC lid.

For fuel assemblies that are shorter than the design basis length, upper and lower fuel spacers (as appropriate) maintain the axial position of the fuel assembly within the MPC basket. The upper fuel spacers are threaded into the underside of the MPC lid as shown in Figure 1.2.5. The lower fuel spacers are placed in the bottom of each fuel basket cell. The upper and lower fuel spacers are designed to withstand normal, off-normal, and accident conditions of storage. An axial clearance of approximately 2 to 2-1/2 inches (5.0 to 6.4 cm) is provided to account for the irradiation and thermal growth of the fuel assemblies. The suggested values for the upper and lower fuel spacer lengths are listed in Tables 2.1.9 and 2.1.10 for each fuel assembly type.

The MPC is constructed entirely from stainless steel alloy materials (except for the neutron absorber and aluminum heat conduction elements). No carbon steel parts are permitted to be in contact with the pool water in the MPC. Concerns regarding interaction of coated carbon steel materials and various MPC operating environments [1.2.1] are not applicable to the MPC. All structural components in a MPC shall be made of Alloy X, a designation which warrants further explanation.

Alloy X is a material which is expected to be acceptable as a Mined Geological Depository System (MGDS) waste package and which meets the thermophysical properties set forth in this document.

Although in practice, type 304 stainless steel has been nearly universally employed for MPCs, for design and analysis purposes, Alloy X (as defined in this FSAR) may be one of the following materials. Any steel part in an MPC may be fabricated from any of the acceptable Alloy X materials listed below, except that the steel pieces comprising the MPC shell (i.e., the 1/2" thick cylinder) must be fabricated from the same Alloy X stainless steel type.

- Type 316
- Type 316LN
- Type 304
- Type 304LN

The Alloy X approach is accomplished by qualifying the MPC for all mechanical, structural, neutronic, radiological, and thermal conditions using material thermophysical properties which are the least favorable for the entire group for the analysis in question. For example, when calculating the rate of heat rejection to the outside environment, the value of thermal conductivity used is the lowest for the candidate material group. Similarly, the stress analysis calculations use the lowest value of the ASME Code allowable stress intensity for the entire group. Stated differently, we have defined a material, which is referred to as Alloy X, whose thermophysical properties, from the MPC design perspective, are the least favorable of the candidate materials.

The evaluation of the Alloy X constituents to determine the least favorable properties is provided in Appendix 1.A.

The Alloy X approach is conservative because no matter which material is ultimately utilized in the MPC construction, the Alloy X approach guarantees that the performance of the MPC will exceed the analytical predictions contained in this document.

1.2.1.2 HI-STAR 100 Overpack

The HI-STAR 100 overpack is a heavy-walled steel cylindrical vessel. Figure 1.2.7 provides a cross sectional elevation view of the HI-STAR 100 overpack. The overpack helium retention boundary is formed by an inner shell welded at the bottom to a cylindrical forging and, at the top, to a heavy main flange with a bolted closure plate. Two concentric grooves are machined into the closure plate for self-energizing seals. The closure plate is recessed into the top flange and the bolted joint is configured to provide maximum protection to the closure bolts and seals in the event of a drop accident. The closure plate has a vent port which is sealed by a threaded port plug with a seal. The bottom plate has a drain port which is sealed by a threaded port plug with a seal. The inner surfaces of the HI-STAR overpack form an internal cylindrical cavity for housing the MPC.

As shown in Figure 1.2.8, the outer surface of the overpack inner shell is buttressed with intermediate shells of gamma shielding which are installed in a manner to ensure a permanent state of contact between adjacent layers. Besides serving as an effective gamma shield, these layers provide additional strength to the overpack to resist potential punctures or penetrations from external missiles. Radial channels are vertically welded to the outside surface of the outermost intermediate shell at nominally equal intervals around the circumference. These radial channels act as fins for improved heat conduction to the overpack outer enclosure shell surface and as cavities for retaining and protecting the neutron shielding. The enclosure shell is formed by welding enclosure shell panels between each of the channels to form additional cavities. Neutron shielding material is placed into each of the radial cavity segments formed by the radial channels, the outermost intermediate shell, and the enclosure shell panels. The exterior flats of the radial channels and the enclosure shell panels form the overpack outer enclosure shell. Atop the outer enclosure shell, rupture disks are positioned in a recessed area. The rupture disks relieve internal pressure which may develop as a result of the fire accident and subsequent off-gassing of the neutron shield material. Within each radial channel, a layer of silicone sponge is positioned, if required to alleviate the thermal stresses from differential expansion. Appendix 1.C provides material information on the thermal expansion foam. Figure 1.2.9 contains a mid-plane cross section of the overpack depicting the inner shell, intermediate shells, radial channels, outer enclosure shell, and neutron shield.

The exposed steel surfaces of the overpack are coated with paint to prevent corrosion. The paint is specified on the design drawings and the material data on the paint is provided in Appendix 1.C. The inner cavity of the overpack is coated with a paint appropriate to its higher temperatures and the exterior of the overpack is coated with a paint appropriate for fuel pool operations and environmental exposure.

Lifting trunnions are attached to the overpack top flange forging for lifting and for rotating the cask body between vertical and horizontal positions. The lifting trunnions are located 180º apart in the sides of the top flange. Pocket trunnions, if used, are welded to the lower side of the overpack to provide a pivoting axis for rotation. The pocket trunnions are located slightly off-center to ensure the proper rotation direction of the overpack. As shown in Figure 1.2.7, the lifting trunnions do not protrude beyond the cylindrical envelope of the overpack enclosure shell. This feature reduces the potential for a direct impact on a trunnion in the event of an overpack side impact.

1.2.1.3 Shielding

The HI-STAR 100 System is provided with sufficient shielding to ensure that the external radiation requirements in 10CFR72.126, 10CFR72.104, and 10CFR72.106 are met. This shielding is an important factor in minimizing personnel doses from gamma and neutron sources in the spent nuclear fuel for ALARA considerations during loading, handling, and storage operations.

The initial attenuation of gamma and neutron radiation emitted by the radioactive spent fuel is provided by the fuel basket structure built from inter-welded intersecting plates and neutron poison panels attached to the fuel storage cell walls. The MPC canister shell, baseplate, and lid provide additional thicknesses of steel to further reduce gamma radiation and, to a smaller extent, neutron radiation at the outer MPC surfaces.

The primary HI-STAR 100 shielding is located in the overpack and consists of neutron shielding and additional layers of steel for gamma shielding. Neutron shielding is provided around the outer circumferential surface of the overpack. Gamma shielding is provided by the overpack inner, intermediate, and enclosure shells with additional axial shielding provided by the bottom plate and the closure plate.

1.2.1.3.1 Neutron Absorber

Boral is a thermal neutron poison material composed of boron carbide and aluminum. Holtec report HI-2033067 [1.2.2] provides a critical appraisal of Boral as a neutron absorber.

Boral was exclusively used in fuel storage applications as the neutron absorber until ca. 2004 when operating problems, particularly hydrogen generation during fuel loading [1.2.2] led to the switch to Metamic [1.2.3, 1.2.4] which is a fully dense metal matrix composite material made of aluminum alloy and boron carbide. A synoptic description of Metamic for use in Holtec fuel baskets has been approved by the USNRC in HI-STORM 100 [1.2.5] as well as in HI-STORM FW [1.2.7], HI-STAR 60 [1.2.8], HI-STORM UMAX [1.2.9] and numerous 10CFR50 dockets. Since 2005, Metamic has been exclusively used in all stainless steel fuel baskets in Holtec's storage and transport systems certified by the USNRC and several other regulatory authorities. Because of Metamic's well-established safety case described in numerous Holtec FSARs and NRC's SERs, the basis for using Metamic is not repeated in this FSAR.

Operating experience in nuclear plants with fuel loading of Boral equipped MPCs as well as laboratory test data indicate that the aluminium used in the manufacture of the Boral may react with water, resulting in the generation of hydrogen. The numerous variables (i.e., aluminium particle size, pool temperature, pool chemistry, etc.) that influence the extent of the hydrogen produced make it impossible to predict the amount of hydrogen that may be generated during MPC loading or unloading at a particular plant. Therefore, due to the variability in hydrogen generation from the Boral-water reaction, the operating procedures in Chapter 8 require monitoring for combustible gases and either exhausting or purging the space beneath the MPC lid during loading and unloading operations when an ignition event could occur (i.e., when the space beneath the MPC lid is open to the welding or cutting operation).

To preclude the incidence of hydrogen generation during MPC welding operation, METAMIC has supplanted Boral in all Holtec dry and wet storage applications since the mid-2000s

1.2.1.3.2 Holtite™ Neutron Shielding

The specification of the overpack neutron shield material is predicated on functional performance criteria. These criteria are:

- Attenuation of neutron radiation and associated neutron capture to appropriate levels;
- Durability of the shielding material under normal conditions, in terms of thermal, chemical, mechanical, and radiation environments;
- Stability of the homogeneous nature of the shielding material matrix;
- Stability of the shielding material in mechanical or thermal accident conditions to the desired performance levels; and
- Predictability of the manufacturing process under adequate procedural control to yield an in-place neutron shield of desired function and uniformity.

Other aspects of a shielding material, such as ease of handling and prior nuclear industry use, are also considered, within the limitations of the main criteria. Final specification of a shield material is a result of optimizing the material properties with respect to the main criteria, along with the design of the shield system, to achieve the desired shielding results.

Holtite-A fulfills the aforementioned criteria in full measure. Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer. Holtite-A is specified with a nominal B_4C loading of 1 weight percent for the HI-STAR 100 System. Appendix 1.B provides the Holtite-A material properties germane to its function as a neutron shield. Holtec has performed confirmatory qualification tests on Holtite-A under the company's QA program.

In the following, a brief summary of the performance characteristics and properties of Holtite-A is provided.

Density

The specific gravity of Holtite-A is 1.68 g/cm^3 as specified in Appendix 1.B. To conservatively bound any potential weight loss at the design temperature and any inability to reach the theoretical density, the density is reduced by 4% to 1.61 g/cm^3. The density used for the shielding analysis is conservatively assumed to be 1.61 g/cm^3 to underestimate the shielding capabilities of the neutron shield.

Hydrogen

The weight concentration of hydrogen is 6.0%. However, all shielding analyses conservatively assume 5.9% hydrogen by weight in the calculations.

Boron Carbide

Boron carbide dispersed within Holtite-A in finely dispersed powder form is present in 1% weight concentration. Holtite-A may be specified with a B_4C content of up to 6.5 weight percent. For the HI-STAR 100 System, Holtite-A is specified with a nominal B_4C weight percent of 1%.

Design Temperature

The design temperature of Holtite-A is set at 300ºF (149°C). The maximum spatial temperature of Holtite-A under all normal operating conditions must be demonstrated to be below this design temperature.

Thermal Conductivity

It is evident from Figure 1.2.9 that Holtite-A is directly in the path of heat transmission from the inside of the overpack to its outside surface. For conservatism, however, the design basis thermal conductivity of Holtite-A under heat rejection conditions is set equal to zero. The reverse condition occurs under a postulated fire event when the thermal conductivity of Holtite-A aids in the influx of heat to the stored fuel in the fuel basket. The thermal conductivity of Holtite-A is conservatively set at 1 Btu/hr-ft-ºF (1.731 W/m°C) for all fire event evaluations.

The Holtite-A neutron shielding material is stable below the design temperature for long-term use and provides excellent shielding properties for neutrons.

1.2.1.3.3 Gamma Shielding Material

For gamma shielding, HI-STAR 100 utilizes carbon steel in plate stock form. Instead of utilizing a thick forging, the gamma shield design in the HI-STAR 100 overpack borrows from the concept of layered vessels from the field of ultra-high pressure vessel technology. The shielding is made from successive layers of plate stock. The fabrication of the shell begins by rolling the inner shell plate and making the longitudinal weld seam. Each layer of the intermediate shells are constructed from two halves. The two halves of the shell shall be precision sheared, bevelled, and rolled to the required radii. The two halves of the second layer are wrapped around the first shell. Each shell half is positioned in its location and while applying pressure using a specially engineered fixture, the halves are tack welded. The bevelled edges to be joined will be positioned to make contact or have a slight root gap. The second layer is made by joining the two halves using two longitudinal welds. Successive layers are assembled in a like manner. Thus, the welding of every successive shell provides a certain inter-layer contact (Figure 1.2.8). The longitudinal and circumferential welds of the intermediate shells are offset from the previous layer, as shown on the drawings in Section 1.5. A thick structural component radiation barrier is thus constructed with four key features, namely:

- The number of layers can be increased as necessary to realize the required design objectives.

- The layered construction is ideal to stop propagation of flaws.

- The thinner plate stock is much more ductile than heavy forgings.

- Post-weld heat treatment is not required by the ASME Code, simplifying fabrication.

1.2.1.4 Lifting Devices

The HI-STAR 100 overpack is equipped with two lifting trunnions located in the top flange. The trunnions are manufactured from a high strength alloy and are installed in tapped openings. The lifting trunnions are designed in accordance with NUREG-0612 and ANSI N14.6. The trunnions are secured in position by a locking pad shaped to make conformal contact with the curved overpack. Once the locking pad is bolted in position, the locking pad inner diameter is sized to restrain the trunnion from backing out.

The lifting, upending, and downending of the HI-STAR 100 System requires the use of external handling devices. A lift yoke is utilized when the cask is to be lifted or set in a vertical orientation. Rotation cradles provide rotation trunnions which interface with pocket trunnions to provide a pivot axis. The lift yoke is connected to the lifting trunnions and the crane hook is used for upending or downending the HI-STAR 100 System by rotating on the rear pocket trunnions. Alternatively, the HI-STAR cask is rotated using a custom designed “rotator cradle,” which does not rely on the pocket trunnions. The pocket trunnions were made optional in the HI-STAR 100 design after early loading experience indicated the pocket trunnions locations to be a localized dose accretors.

The top of the MPC lid is equipped with four threaded holes that allow lifting of the loaded MPC. These holes allow the loaded MPC to be raised/lowered from the HI-STAR overpack. MPC handling operations are performed using a HI-TRAC transfer cask of the HI-STORM 100 System (Docket No. 72-1014). The HI-TRAC transfer cask allows the sealed MPC loaded with spent fuel to be transferred from the HI-STORM Overpack (storage-only) to the HI-STAR Overpack, or vice versa. The threaded holes in the MPC lid are designed in accordance with NUREG-0612 and ANSI N14.6.

1.2.1.5 Design Life & Service Life

The design life of the HI-STAR 100 System in long term storage is given in Table 2.0.1. The design life is the length of time for which the storage system has been engineered to conservatively render all of its intended design functions. This is accomplished by using materials of construction with a long proven history in the nuclear industry and specifying materials known to withstand their operating environments with little to no degradation. A maintenance program, as specified in Chapter 9, is also implemented to ensure the HI-STAR 100 System will exceed its design life. . Because there are no identifiable failure modes that may limit the active useable life of the HI-STAR 100 cask (also referred to as the “service life”), the expected service life has been set as 100 years. The design features and considerations that assure the HI-STAR 100 System will perform as designed throughout the service life include the following:

<u>HI-STAR Overpack</u>

- Exposure to Environmental Effects
- Material Degradation

- Maintenance and Inspection Provisions

MPC

- Corrosion
- Structural Fatigue Effects
- Maintenance of Helium Atmosphere
- Allowable Fuel Cladding Temperatures
- Neutron Absorber Boron Depletion

The adequacy of the HI-STAR 100 System for its design life is discussed in sub-sections 3.4.10 and 3.4.11.

1.2.1.6 Support Structure for Horizontal Casks

In the horizontal orientation for storage, the HI-STAR 100 is supported by saddle-type supports located near the cask's extremities as shown in Figure 1.2.13. As can be seen in the figure, the saddles support the cask from below. Steel tie-down straps wrap around the remainder of the circumference at the same axial locations to secure the cask to the saddles. A tilting plate bolted to the cask base surface assists in rotating the cask from vertical to horizontal, and is left in place afterwards to provide additional gamma and neutron shielding

Basic dimensions of the saddle supports and tie-down straps are presented in Table 1.2.7. These saddle dimensions shall be used whenever the HI-STAR 100 cask is held horizontally.

1.2.2 Operational Characteristics

1.2.2.1 Design Features

The HI-STAR 100 System is engineered to store different types of MPCs for varying PWR and BWR fuel characteristics.

The HI-STAR 100 System can safely store spent nuclear fuel with minimum cooling times. The maximum thermal decay heat load and SNF enrichments for each of the MPCs are identified in Chapter 2. The decay heat emitted by the spent nuclear fuel is dissipated in an entirely passive mode without any mechanical or forced cooling.

Both the free volume of the HI-STAR 100 MPCs and the annulus between the external surface of the MPC and the inside surface of the overpack are inerted with commercially pure helium gas during the spent nuclear fuel loading operations. Table 1.2.2 specifies the helium pressure to be placed in the MPC internal cavity.

The primary heat transfer mechanisms are metal conduction and surface radiation for the HI-STAR 100 System. The MPC internal helium atmosphere, in addition to providing a noncorrosive dry atmosphere for the fuel cladding, provides for heat transfer through helium conduction. The most adverse temperature profiles and thermal gradients for the HI-STAR 100 System with each of the

MPCs are discussed in detail in Chapter 4.

The criticality control features of the HI-STAR 100 are designed to maintain the neutron multiplication factor k-effective (including uncertainties and calculational bias) at less than 0.95 under all normal, off-normal, and accident conditions of storage as analyzed in Chapter 6.

1.2.2.2 Sequence of Operations

Table 1.2.6 provides the basic sequence of operations necessary to defuel a spent fuel pool using the HI-STAR 100 System. The detailed sequence of steps for storage-related loading and handling operations is provided in Chapter 8 and is supported by the drawings in Section 1.5. A summary of general actions needed for the loading and unloading operations is provided below.

Loading Operations

At the start of loading operations, the overpack is configured with the closure plate removed. The lift yoke is used to position the overpack in the designated preparation area or setdown area for overpack inspection and MPC insertion. The annulus is filled with plant demineralized water and an inflatable annulus seal is installed. The inflatable seal prevents contact between spent fuel pool water and the MPC shell reducing the possibility of contaminating the outer surfaces of the MPC. The MPC is then filled with spent fuel pool water or plant demineralized water. The overpack and MPC are lowered into the spent fuel pool for fuel loading using the lift yoke. Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed.

While still underwater, a thick shielding lid (the MPC lid) is installed. The lift yoke is remotely engaged to the overpack lifting trunnions and is used to lift the overpack close to the spent fuel pool surface. As an ALARA measure, dose rates are measured on the top of the overpack and MPC prior to removal from the pool to check for activated debris on the top surface. The MPC lift bolts (securing the MPC lid to the lift yoke) are removed. As the overpack is removed from the spent fuel pool, the lift yoke and overpack are sprayed with demineralized water to help remove contamination.

The overpack is removed from the pool and placed in the designated preparation area. The top surfaces of the MPC lid and the top flange of the overpack are decontaminated. The inflatable annulus seal is removed, and an annulus shield is installed. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped into the annulus. Dose rates are measured to ensure that the dose rates are within expected values. The Automated Welding System baseplate shield is installed to reduce dose rates around the top of the cask. The MPC water level is lowered slightly and the MPC lid is seal-welded using the Automated Welding System (AWS). Liquid penetrant examinations are performed on the root and final passes.

A progressive liquid penetrant examination is also performed on the MPC lid-to-shell weld. The MPC water is displaced from the MPC by blowing pressurized helium or nitrogen gas into the vent port of the MPC, thus displacing the water through the drain line. At the appropriate time in the sequence of activities, based on the type of test perform (hydrostatic or pneumatic), a pressure test of the MPC enclosure vessel is performed.

The Vacuum Drying System (VDS) is connected to the MPC and is used to remove all residual water from the MPC in a stepped evacuation process. The stepped evacuation process is used to preclude the formation of ice in the MPC and VDS lines. The internal pressure is reduced and held for a duration to ensure that all liquid water has evaporated.

Following this dryness test, the VDS is disconnected, the Helium Backfill System (HBS) is attached, and the MPC is backfilled with a predetermined amount of helium gas. The helium backfill ensures adequate heat transfer during storage, provides an inert atmosphere for long-term fuel integrity, and provides the means of future leakage rate testing of the MPC confinement boundary welds. Cover plates are installed and seal-welded over the MPC vent and drain ports with liquid penetrant examinations performed on the root and final passes. The cover plates are helium leakage tested to confirm that they meet the established leakage rate criteria.

The Forced Helium Dehydration system, described in Appendix 2.B of the HI-STORM 100 FSAR can be used in lieu of the Vacuum Drying and Helium Back-fill systems. The use of FHD is preferred if the MPC contents include high burn-up fuel or water-logged fuel.

The MPC closure ring is then placed on the MPC, aligned, tacked in place, and seal welded, providing redundant closure of the MPC confinement cavity closure welds. Tack welds are visually examined, and the root and final welds are inspected using the liquid penetrant examination technique to ensure weld integrity. The annulus shield is removed and the remaining water in the annulus is drained. The AWS Baseplate shield is removed. The MPC lid and accessible areas of the top of the MPC shell are smeared for removable contamination and overpack dose rates are measured. The overpack closure plate is installed and the bolts are torqued. The overpack annulus is dried using the VDS, and backfilled with helium gas for heat transfer and seal testing. Concentric metallic seals in the overpack closure plate prevent the leakage of the helium gas from the annulus and provide an additional confinement boundary to the release of radioactive materials. The seals on the overpack vent and drain port plugs are leak tested along with the overpack closure plate inner seal. Cover plates with metallic seals are installed over the overpack vent and drain ports to provide redundant closure of the overpack penetrations. A port plug with a metallic seal is installed in the overpack closure plate test port to provide fully redundant closure of all potential leakage paths in the overpack penetrations.

The overpack is secured to the transporter and moved to the ISFSI pad (the term ISFSI pad is used in the generic sense to mean the location where the cask will be staged for temporary or extended storage).

The HI-STAR 100 System can also be remotely loaded at a specially-designed dry loading facility (i.e., hot cell) with appropriate modifications to the loading procedures.

Unloading Operations

The HI-STAR 100 System unloading procedures describe the general actions necessary to prepare the MPC for unloading, cool the stored fuel assemblies in the MPC, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover the overpack and empty MPC. Special precautions are outlined to ensure personnel safety during the unloading operations, and to prevent the risk of MPC overpressurization and thermal shock to the stored spent fuel assemblies.

The overpack and MPC are returned to the designated preparation area from the ISFSI. At the site's discretion, a gas sample is drawn from the annulus and analyzed. The gas sample provides an indication of MPC confinement performance. The annulus is depressurized, the overpack closure plate is removed, and the annulus is filled with plant demineralized water. The annulus and overpack top surface are protected from debris that will be produced by removing the MPC lid.

The MPC closure ring and vent and drain port cover plates are core drilled. Local ventilation is established round the MPC ports. The RVOAs are attached to the vent and drain ports. The RVOAs allow access to the inner cavity of the MPC, while providing a hermetic seal. The MPC is cooled using a closed loop heat exchanger, if necessary to reduce the MPC internal temperature to allow water flooding. Following fuel cooldown, the MPC is flooded with water. The MPC lid-to-shell weld is removed. Then all weld removal equipment is removed with the MPC lid left in place.

The inflatable annulus seal is installed and pressurized. The MPC lid is rigged to the lift yoke and the lift yoke is engaged to overpack lifting trunnions. The overpack is placed in the spent fuel pool and the MPC lid is removed. All fuel assemblies are returned to the spent fuel storage racks and the MPC fuel cells are vacuumed to remove any assembly debris. The overpack and MPC are returned to the designated preparation area where the MPC water is pumped back into the spent fuel pool. The annulus water is drained and the MPC and overpack are decontaminated in preparation for re-utilization.

The HI-STAR 100 System can also be remotely unloaded at a specially designed dry unloading facility (i.e., hot cell) with appropriate modifications to the unloading procedures.

1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

1.2.2.3.1 Criticality Prevention

Criticality is controlled by geometry and neutron absorption materials in the fuel basket. The MPC-24 and MPC-32 rely on soluble boron credit in the MPC water during wet loading and unloading operations. The MPC-24 may be used for PWR fuel without taking any soluble boron credit if the enrichment limit is met.

1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the HI-STAR 100 dry storage system. A detailed evaluation is provided in Section 3.4.

1.2.2.3.3 Operation Shutdown Modes

The HI-STAR 100 System is totally passive and consequently, operation shutdown modes are unnecessary. Guidance is provided in Chapter 8, which outlines the HI-STAR 100 unloading procedures, and Chapter 11, which outlines the corrective course of action in the wake of all postulated accidents.

1.2.2.3.4 Instrumentation

As stated earlier, the HI-STAR 100 confinement boundary is the MPC, which is seal welded, progressive liquid penetrant examined, pressure tested, and leak tested. Including the overpack, there are three completely independent barriers to the release of radioactivity to the outside environment. These barriers, proven through decades of use in numerous industries, are arrayed in a sequential manner, making the escape of radioactivity to the outside environment unlikely. The HI-STAR 100 is a completely passive system with appropriate margins of safety; therefore, it is not necessary to deploy any instrumentation to monitor the cask in the storage mode, and none is provided.

1.2.2.3.5 Maintenance Technique

Because of their passive nature, the HI-STAR 100 Systems require minimal maintenance over their lifetime. Chapter 9 describes the acceptance criteria and maintenance program set forth for the HI-STAR 100 System.

1.2.3 Cask Contents

The HI-STAR 100 System is designed to house different types of MPCs. The MPCs are designed to store both BWR and PWR spent nuclear fuel assemblies. Tables 1.2.1 and 1.2.2 provide key design parameters for the MPCs. A description of acceptable fuel assemblies including damaged fuel and fuel debris for storage in the MPCs is provided in Chapter 2.

Up to 4 damaged fuel containers containing specified fuel debris may be stored within an MPC-68F.

Table 1.2.1

KEY SYSTEM DATA FOR HI-STAR 100
(See Chapter 2 for limitation on damaged fuel and fuel debris permitted in various MPC types)

	QUANTITY	NOTES
Types of MPCs included in this revision of the submittal	3	2 for PWR 1 for BWR
MPC storage capacity:	MPC-24	Up to 24 intact zircaloy or stainless steel clad PWR fuel assemblies
	MPC-68	Up to 68 intact zircaloy or intact stainless steel clad BWR fuel assemblies or damaged zircaloy clad fuel assemblies in damaged fuel containers in the MPC-68 or Up to 4 damaged fuel containers with zircaloy clad BWR fuel debris and the complement intact or damaged zircaloy clad BWR fuel assemblies within an MPC-68F.
	MPC-32	Up to 32 intact zircaloy clad PWR fuel assemblies

Table 1.2.2

KEY PARAMETERS FOR HI-STAR 100 MULTI-PURPOSE CANISTERS

<table>
<tr><th></th><th colspan="2">PWR</th><th colspan="2">BWR</th></tr>
<tr><td>Pre-disposal service life, years</td><td colspan="2">50</td><td colspan="2">50</td></tr>
<tr><td>Design temperature, ºF (°C)
Max
Min</td><td colspan="2">
752 ºF (400°C)
-40 °F (-40 ºC)</td><td colspan="2">
752 ºF (400°C)
-40 °F (-40 ºC)</td></tr>
<tr><td>Design internal pressure, psig (kpad)

Normal conditions
Off-normal conditions
Accident Conditions</td><td colspan="2">

100 (690)
100 (690)
200 (1379)</td><td colspan="2">

100 (690)
100 (690)
200 (1379)</td></tr>
<tr><td>Total heat load, max. (kW)
Vertical Orientation
Horizontal Orientation</td><td>MPC-24
19 kW
20 kW</td><td>MPC-32
18.5 kW
20 kW</td><td colspan="2">
18.5 (MPC-68)
18.5 (MPC-68)</td></tr>
<tr><td>Maximum permissible peak fuel cladding temperature:
Normal
Short Term & Accident</td><td colspan="2">See Chapter 4</td><td colspan="2">See Chapter 4</td></tr>
<tr><td>MPC internal environment Helium fill at 70°F (21°C), psig (kpad)
Vertical Orientation
Horizontal Orientation</td><td colspan="2">
≤ 22.2 (153)
≤ 44.8 (309)</td><td colspan="2">
≤ 22.2 (153)
≤ 44.8 (309)</td></tr>
<tr><td>MPC external environment/overpack internal pressure

Helium fill initial pressure, psig (kpad) at STP</td><td colspan="2">10 (69)</td><td colspan="2">10 (69)</td></tr>
<tr><td>Maximum permissible reactivity including all uncertainties and biases</td><td colspan="2"><0.95</td><td colspan="2"><0.95</td></tr>
<tr><td>Fixed Neutron Absorber ^{10}B Areal Density (g/cm^2)
Boral
Metamic</td><td>MPC-24

0.0267
0.0223</td><td>MPC-32

0.0372
0.0310</td><td>MPC-68

0.0372
0.0310</td><td>MPC-68F
(Note 1)
0.01
NA</td></tr>
<tr><td>End closure(s)</td><td colspan="2">Welded</td><td colspan="2">Welded</td></tr>
</table>

	PWR	BWR
Fuel handling	Opening compatible with standard grapples	Opening compatible with standard grapples
Heat dissipation	Passive	Passive

NOTE:
1. All MPC-68F canisters are equipped with Boral neutron absorber.

Table 1.2.3

INTENTIONALLY DELETED

Table 1.2.4

INTENTIONALLY DELETED

Table 1.2.5

INTENTIONALLY DELETED

Table 1.2.6

HI-STAR 100 OPERATIONS DESCRIPTION

Site-specific handling and operations procedures will be prepared, reviewed, and approved by each owner/user.	
1	Overpack and MPC lowered into the fuel pool without closure plate and MPC lid
2	Fuel assemblies transferred into the MPC fuel basket
3	MPC lid lowered onto the MPC
4	Overpack/MPC assembly moved to the decon pit and MPC lid welded in place, progressive liquid penetrant examined, and pressure tested.
5	MPC dewatered, vacuum dried, backfilled with helium and the vent/drain port cover plates and closure ring welded (The Forced Helium Dehydrator may be used in lieu of the VDS)
6	Overpack drained and external surfaces decontaminated
7	Overpack seals and closure plate installed and bolts torqued
8	Overpack cavity dried, backfilled with helium, and helium leak tested
9	HI-STAR 100 loaded onto transporter and moved to the ISFSI pad for on-site storage
10	HI-STAR 100 emplaced onto the ISFSI pad at its designated location

TABLE 1.2.7

DIMENSIONS OF SUPPORT STRUCTURE FOR HORIZONTAL-ORIENTATION CASKS

Dimension	S.I Value	U.S. Value
Saddle Center-to-Center Spacing	3632 mm	143 in
Saddle Radius	1226 mm	48 9/32 in
Saddle Arc Length	140 deg.	140 deg.
Saddle Width	290 mm	11 13/32 in
Cask Centerline Height	1780 mm	70 1/8 in
Tie-Down Strap Width	230 mm	9 1/16 in
Tie-Down Strap Radius	1244 mm	48 3/16 in
Tie-Down Strap Thickness	25 mm	31/32 in

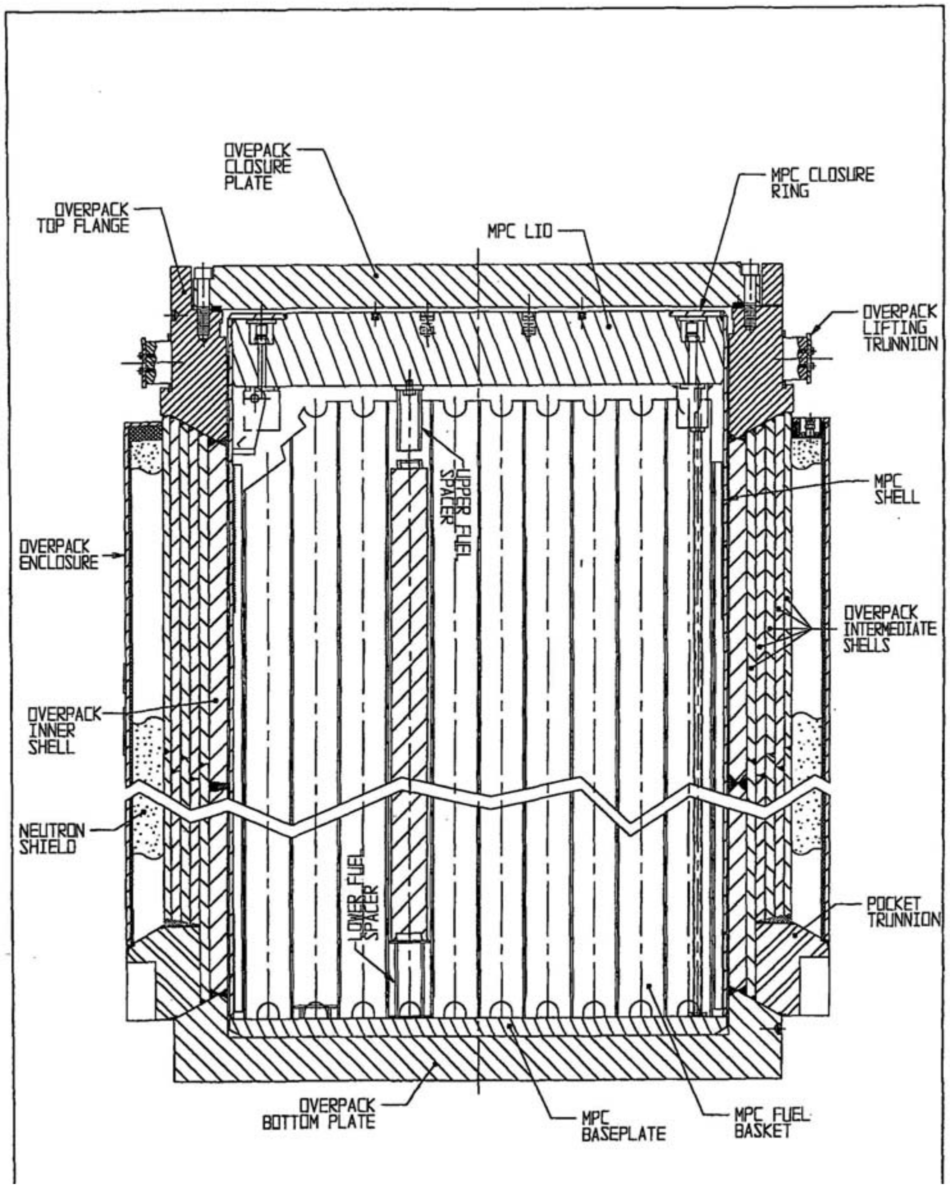


Figure 1.2.1; CROSS SECTION ELEVATION VIEW OF HI-STAR 100 SYSTEM

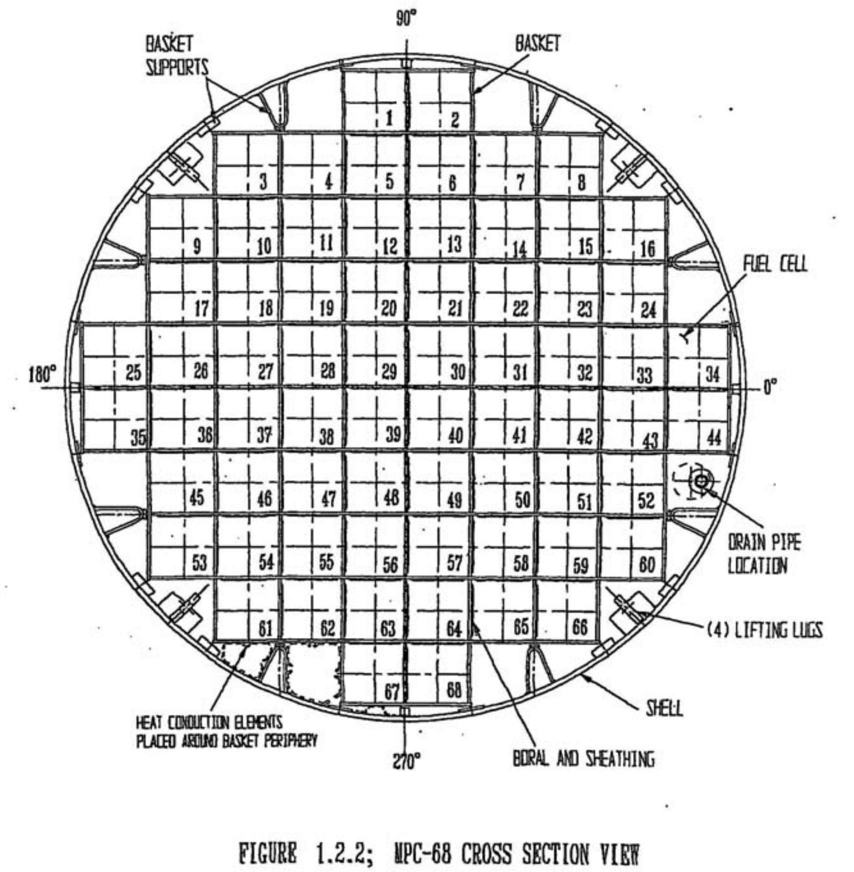


FIGURE 1.2.2; MPC-68 CROSS SECTION VIEW

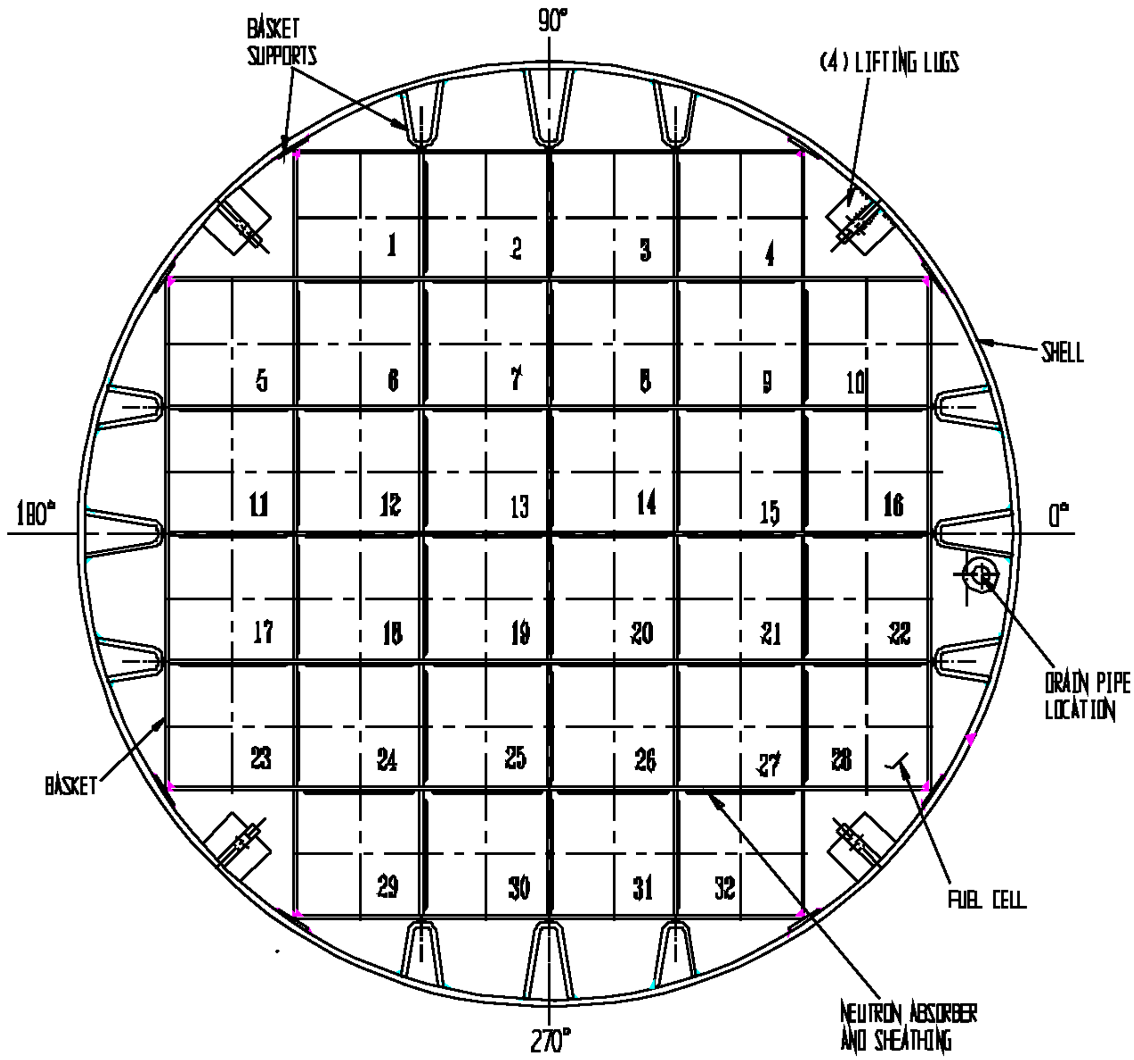


FIGURE 1.2.3; MPC-32 CROSS SECTION

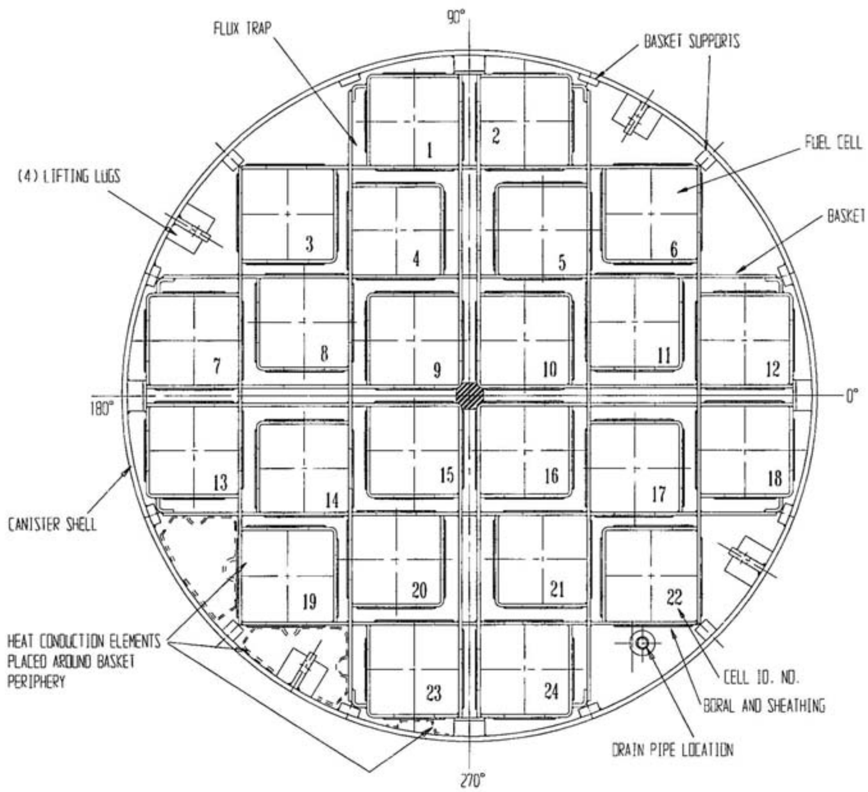


FIGURE 1.2.4; MPC-24 CROSS SECTION VIEW

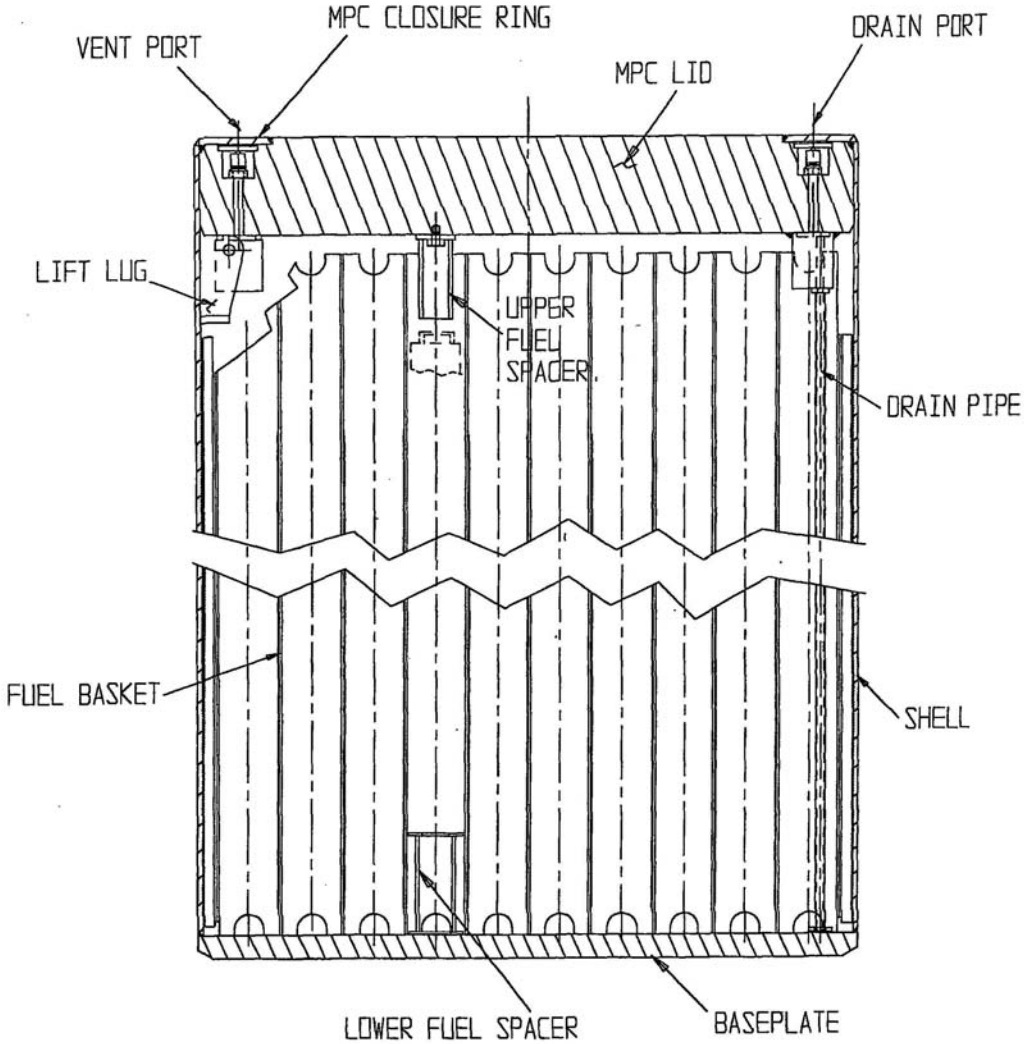


FIGURE 1.2.5; CROSS SECTION ELEVATION VIEW OF MPC

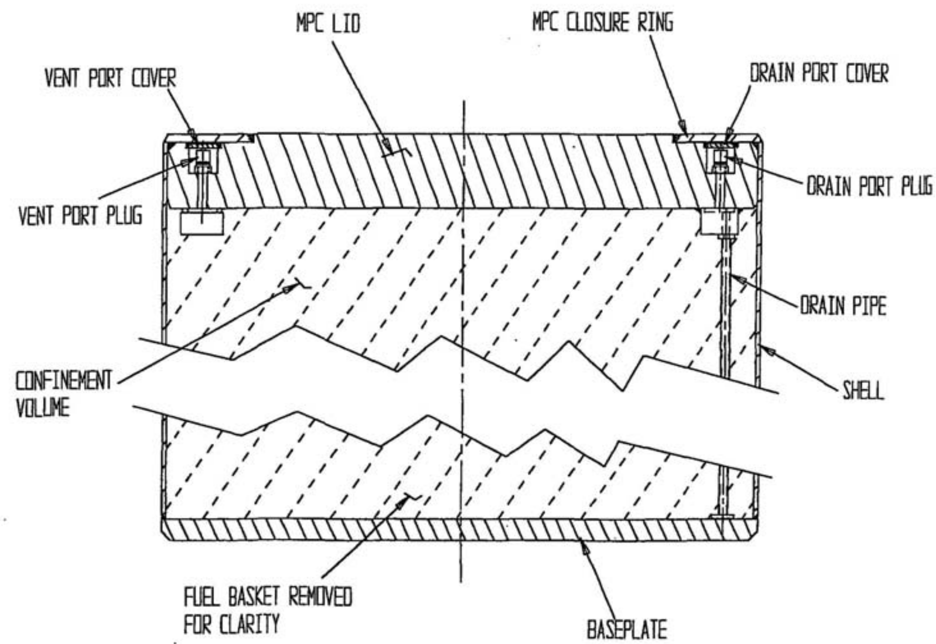


FIGURE 1.2.6; MPC CONFINEMENT BOUNDARY

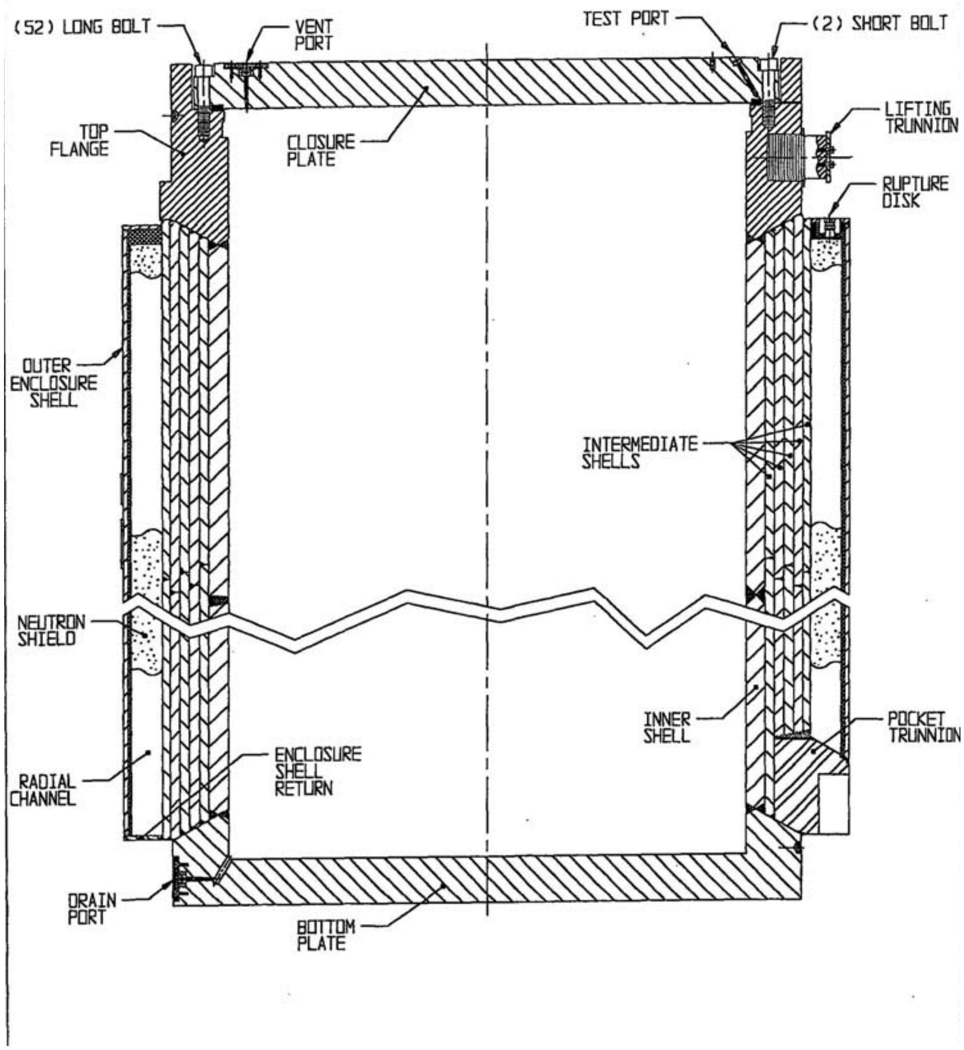
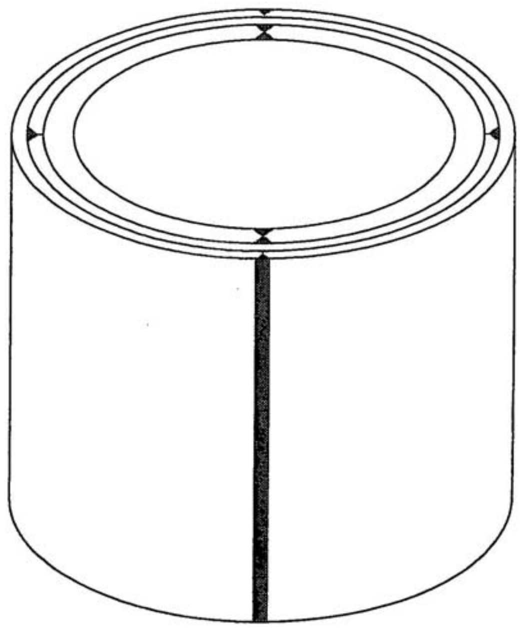
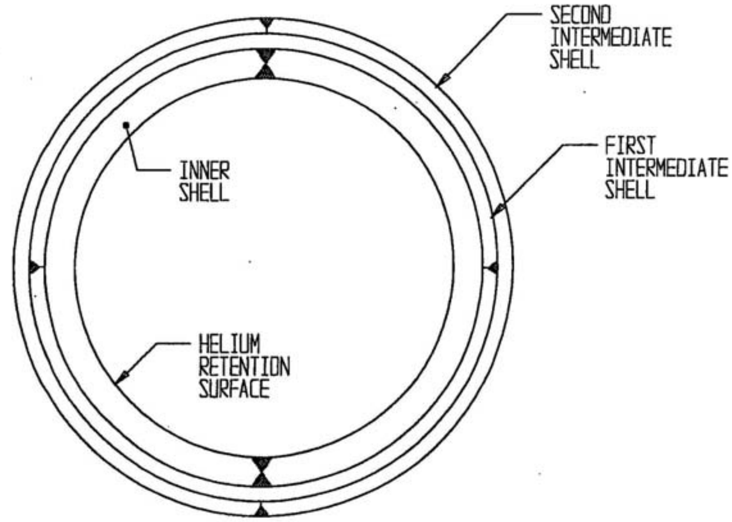


FIGURE 1.2.7; CROSS SECTION ELEVATION VIEW OF OVERPACK



ISOMETRIC VIEW OF CENTRAL REGION OF THE OVERPACK



CROSS SECTION AT MID-HEIGHT

FIGURE 1.2.8; HI-STAR 100 OVERPACK SHELL LAYERING

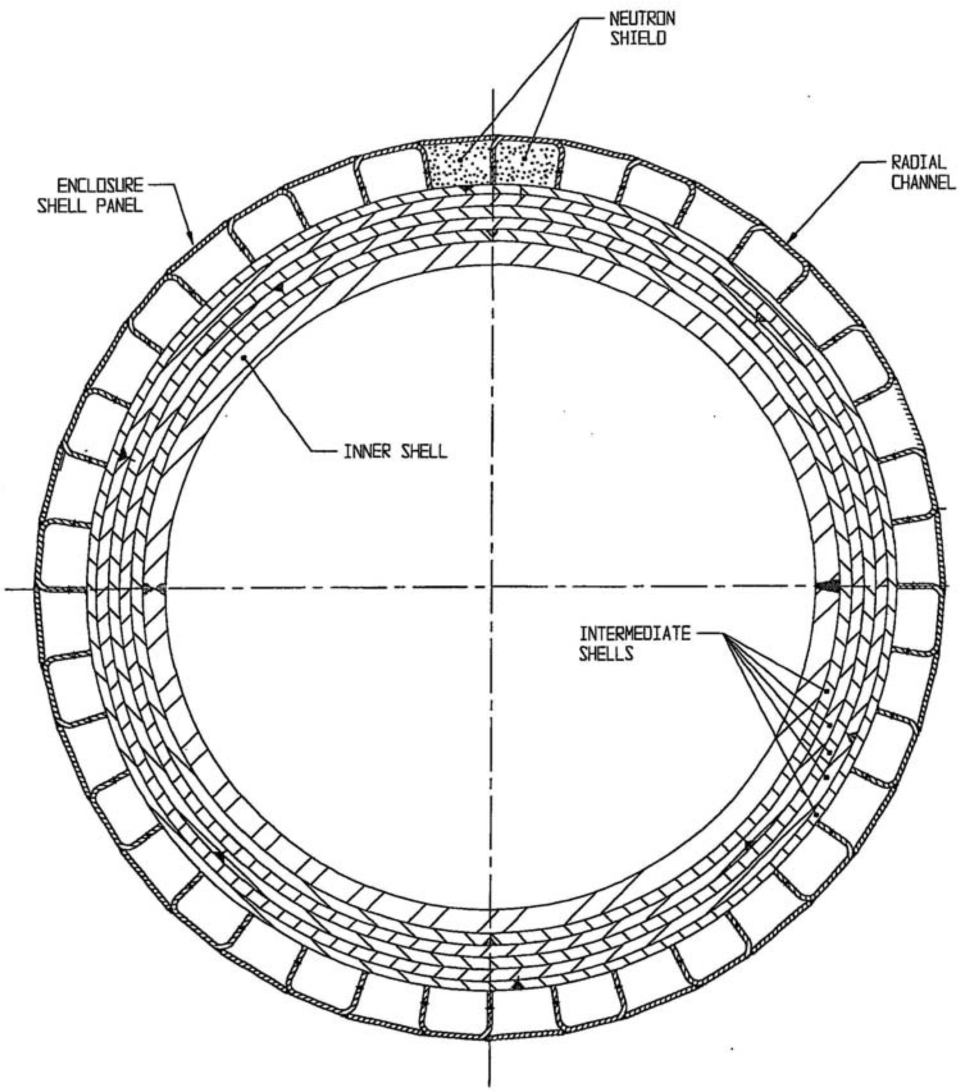


FIGURE 1.2.9; OVERPACK MID-PLANE CROSS SECTION

FIGURE 1.2.10

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Figures 1.2.11a – 1.2.12c

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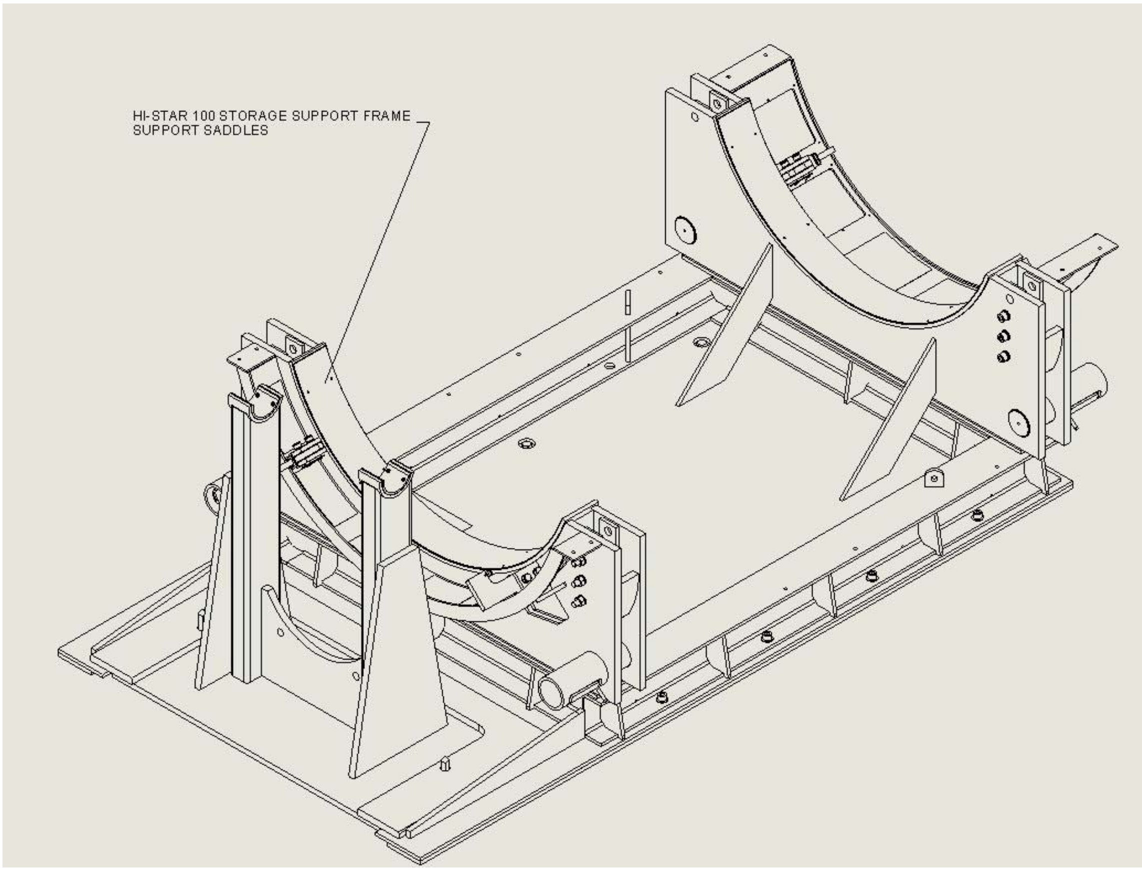
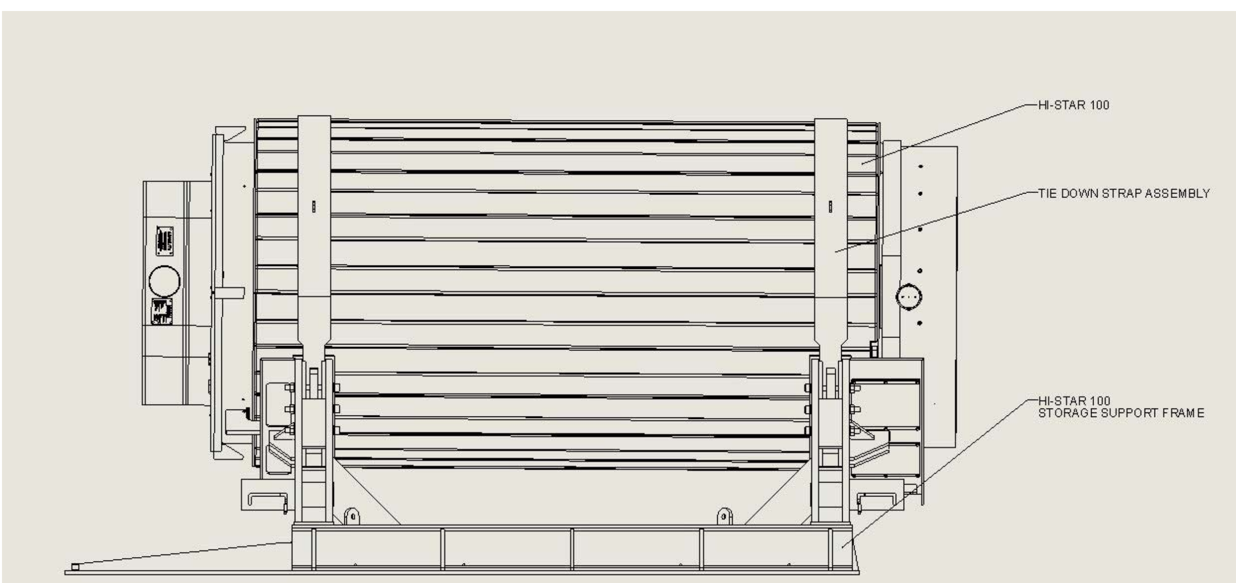
FIGURE 1.2.13

SUPPORT STRUCTURE FOR HORIZONTAL-ORIENTATION CASKS





1.3 IDENTIFICATION OF AGENTS AND CONTRACTORS

This section contains the necessary information to fulfill the requirements pertaining to the qualifications of the applicant pursuant to 10 CFR72.2(a)(1),(b) and 72.230(a). Holtec International (www.Holtecinternational.com) is the system designer and applicant for certification of the HI-STAR 100 system.

Holtec International is an engineering technology company with a principal focus on the power industry. Holtec International Nuclear Power Division (NPD) specializes in spent fuel storage technologies. NPD has carried out turnkey wet storage capacity expansions (engineering, licensing, fabrication, removal of existing racks, performance of underwater modifications, volume reduction of the old racks and hardware, installation of new racks, and commissioning of the fuel pool for increased storage capacity) in numerous nuclear plants around the world. Over 90 plants in the U.S., Britain, Brazil, Korea, Mexico, China and Taiwan have utilized the Company's wet storage technology to establish their state-of-the-art in-pool storage capacities.

Holtec's NPD is also a turnkey provider of dry storage and transportation technologies to nuclear plants around the globe. The company is contracted by over 50 nuclear units in the U.S. to provide the company's vertical ventilated dry storage technology. Utilities in China, Korea, Spain, Ukraine, Belgium, Sweden; the United Kingdom and Switzerland are also active users of Holtec International's dry storage and transport systems.

Four U.S. commercial plants, namely, Dresden Unit 1, Trojan, Indian Point Unit 1, and Humboldt Bay have thus far been completely defueled using Holtec International's technology. For many of its dry storage clients, Holtec International provides all phases of dry storage including: the required site-specific safety evaluations; ancillary designs; manufacturing of all capital equipment; preparation of site construction procedures; personnel training; dry runs; and fuel loading. The USNRC dockets in parts 71 and 72 currently maintained by the Company are listed in Table 1.3.1.

Holtec International's corporate engineering consists of professional engineers and experts with extensive experience in every discipline germane to the fuel storage technologies, namely structural mechanics, heat transfer, computational fluid dynamics, and nuclear physics. Virtually all engineering analyses for Holtec's fuel storage projects (including HI-STAR 100) are carried out by the company's full-time staff. The Company is actively engaged in a continuous improvement program of the state-of-the-art in dry storage and transport of spent nuclear fuel. The active patents and patent applications in the areas of dry storage and transport of SNF held by the Company (ca. January 2012) are listed in Table 1.3.2. Table 1.3.3 1.3.3 lists Holtec patents on dry storage technologies that have been published by the US patent office (as of Jan .2012) but not yet granted as well as any provisional patents. Many of these listed patents have been utilized in the design of the HI-STAR 100 System.

Holtec International's quality assurance (QA) program was originally developed to meet NRC requirements delineated in 10CFR50 [1.3.1], Appendix B, and was expanded to include provisions of 10CFR71[1.0.4], Subpart H, and 10CFR72, Subpart G, for structures, systems, and components designated as important to safety. The Holtec quality assurance program, which satisfies all 18 criteria in 10CFR72, Subpart G, that apply to the design, fabrication, construction, testing, operation, modification, and decommissioning of structures, systems, and components important to safety is incorporated by reference into this FSAR. Holtec International's QA program has been certified by the USNRC (Certificate No. 71-0784).

The HI-STAR 100 System will be fabricated at Holtec's own manufacturing plants operating under the Company's corporate QA program. If another fabricator is to be used for the fabrication of any part of the HI-STAR 100 System, the proposed fabricator will be evaluated and audited in accordance with Holtec International's QA program approved by the USNRC.

Holtec International's Nuclear Power Division (NPD) also carries out site services for dry storage deployments at nuclear power plants. Several nuclear plants, such as Trojan, Waterford 3 and Comanche Peak have deployed dry storage at their sites using a turnkey contract with Holtec International.

Table 1.3.1

USNRC DOCKETS ASSIGNED TO HOLTEC INTERNATIONAL

System Name	Docket Number
HI-STORM 100 (Storage)	72-1014
HI-STAR 100 (Storage)	72-1008
HI-STAR 100 (Transportation)	71-9261
HI-STAR 180 (Transportation)	71-9325
HI-STAR 60 (Transportation)	71-9336
HI-STAR 180D (Transportation)	71-9367
Holtec Quality Assurance Program	71-0784
HI-STORM FW (Storage)	72-1032
HI-STORM UMAX	72-1040

Table 1.3.2
DRY STORAGE AND TRANSPORT PATENTS HELD BY HOLTEC INTERNATIONAL

No.	Colloquial Name of the Patent	USPTO Patent Number
1.	Honeycomb Fuel Basket	5,898,747
2.	Radiation Absorbing Refractory Composition (METAMIC)	5,965,829
3.	HI-STORM Overpack (all HI-STORM Models)	6,064,710
4.	Extrusion Fabrication Process for Discontinuous Carbide Particulate Metal Matrix Composites and Super Hypereutectic A1/S1(METAMIC-CLASSIC)	6,042,779
5.	Duct Photon Attenuator	6,519,307B1
6.	HI-TRAC Operation	6,587,536B1
7.	Cask Mating Device (Hermetically Sealable Transfer Cask)	6,625,246B1
8.	Improved Ventilator Overpack	6,718,000B2
9.	Below Grade Transfer Facility	6,793,450B2
10.	HERMIT (Seismic Cask Stabilization Device)	6,848,223B2
11.	Cask Mating Device (operation)	6,853,697
12.	Davit Crane	6,957,942B2
13.	Duct-Fed Underground HI-STORM	7,068,748B2
14.	Forced Helium Dehydrator (design)	7,096,600B2
15.	Below Grade Cask Transfer Facility	7,139,358B2
16.	Forced Gas Flow Canister Dehydration (alternate embodiment)	7,210,247B2
17.	HI-TRAC Operation (Maximizing Radiation Shielding During Cask Transfer Procedures)	7,330,525
18.	HI-STORM 100U	7,330,526B2
19.	Flood Resistant HI-STORM	7,590,213B1
20.	HI-STORM 100M (Underground Manifolded module assembly)	7,676,016B2
21.	Method and Apparatus for Dehydrating High Level Waste Based On Dew Point Temperature Measurements	7,707,741B2
22.	Optimized Weight Transfer Cask with Detachable Shielding	7,786,456B2
23.	VECASP (Apparatus, System, and Method for Facilitating Transfer of High Level Radioactive Waste to and/or From a Pool	7,820,870B2
24.	HI-STORM 100F (Counter-flow Underground Vertical Ventilated Module)	7,933,374B2
25.	Apparatus for Transporting and/or Storing Radioactive Materials Having Jacket Adapted to Facilitate Thermo-siphon Fluid Flow	7,994,380B2
26.	Method of Removing Radioactive Materials from Submerged State and/or Preparing Spent Nuclear Fuel for Dry Storage	8,067,659B2
27.	HI-STORM 100US	8,098,790B1
28.	Canister Apparatus and Basket for Transporting, Storing and/or Supporting Spent Nuclear Fuel(Double Wall Canister)	8,135,107B2
29.	System And Method For The Ventilated Storage Of High Level	8,660,230B2

	Radioactive Waste In A Clustered Arrangement(HIC-Storm)	
30.	System And Method For Preparing A Container Loaded With Wet Radioactive Elements For Dry Storage(Inter-Unit Transfer)	8,561,318B2
31.	Apparatus And Method For Supporting Fuel Assemblies In An Underwater Environment Having Lateral Access Loading	8,139,706B2
32.	Fuel Basket Spacer, Apparatus And Method Using The Same For Storing High Level Radioactive Waste (HI-STAR 180)	8,712,001B2
33.	Method And Apparatus For Dehydrating High Level Waste Based On Dew Point Temperature Measurements (FHD)	8,266,823B2
34.	Atomized Picoscale Composite Aluminum Alloy And Method Thereof	8,323,373B2
35.	Apparatus System And Method For Low Profile Translation Of High Level Radioactive Waste Containment Structure (Low Profile Transporter)	8,345,813
36.	Method Of Storing High Level Waste (100F)	8,345,813B2
37.	Apparatus For Providing Additional Radiation Shielding To A Container Holding Radioactive Materials, and Method of Using The Same To Handle And/Or Process Radioactive Materials	8,415,521B2
38.	Spent Fuel Basket, Apparatus and Method Using the Same For Storing High Level Radioactive Waste	8,548,112B2
39.	Apparatus For Supporting Radioactive Fuel Assemblies and Methods of Manufacturing The Same	8,576,976B2
40.	Single-Plate Neutron Absorbing Apparatus And Method Of Manufacturing The Same	8,681.924B2
41.	Method Of Transferring High Level Radioactive Materials, And System For The Same	8,717221B2
42.	Method and Apparatus for Preparing Spent Nuclear Fuel for Dry Storage	8,737,559B2
43.	Apparatus For Storing And/Or Transporting High Level Radioactive Waste, And Method For Manufacturing The Same	8,798,224B2
44.	System and Method for Transferring and/or Working Near a Radioactive Payload Using Shielding-Gate Apparatus	8,884,259B
45.	Ventilated System For Storing High Level Radioactive Waste	8,905,259B2
46.	Canister Apparatus And Basket For Transporting, Storing, And/Or Supporting Spent Nuclear Fuel	8,929,504B2
47.	System, Method And Apparatus For Providing Additional Radiation Shielding to High Level Radioactive Materials	8,995,04B26
48.	System, Method And Apparatus For Providing Additional Radiation Shielding To High Level Radioactive Materials	8,995,604B2
49.	System and Method for Reclaiming Energy from Heat Emanating from Spent Nuclear Fuel	9,001,958B2

Table 1.3.3

HOLTEC INTERNATIONAL PENDING PATENTS ON FUEL STORAGE

	Title	Submittal Date	*USPTO FILE NUMBER*	
1.	Neutron Shielding Ring, Apparatus And Method Using The Same For Storing High Level Radioactive Waste (HI-STAR 180)	02-Jul-07	*11772581*	US20080084958
2.	Spent Fuel Basket, Apparatus And Method Using The Same For Storing High Level Radioactive Waste (HI-STAR 180)	02-Jul-07	*11772610*	US20080031396
3.	System And Method For Storing Spent Nuclear Fuel Having Manifolded Underground Vertical Ventilated Module (100M)	19-Feb-10	*12709094*	US20100150297
4.	Cask Apparatus, System And Method For Transporting And/Or Storing High Level Waste (HI-SAFE)	28-Apr-10	*12769622*	US20100272225
5.	Spent Fuel Basket For Storing High Level Radioactive Waste (HEXCOMB Racks)	29-Oct-08	*12260914*	US20090175404
6.	Container and System for Handling Damaged Nuclear Fuel, and Method of Making the Same	19-Feb-14	*14239752*	WO2013055445
7.	Vertical Ventilated Cask With Distributed Air Inlets For Storing Fissile Nuclear Materials	13-May-14	*14358032*	US2014329455A1
8.	A Method To Control The Temperature Of A Ventilated Spent Nuclear Fuel or Nuclear Waste Storage System By Obstructing the Flow of Cooling Air	28-Apr-14	*14354851*	WO2013085638
9.	System And Method For Storing And Leak Testing A Radioactive Materials Storage Canister	13-Sep-13	*N/A*	WO2014036561
10.	A Method Of Removing Radioactive Materials From A Submerged State And/Or Preparing Spent Nuclear Fuel For Dry Storage	29-Nov-11	*N/A*	US20120142991

1.4 GENERIC CASK ARRAYS

The only system required for storage of the HI-STAR 100 System is the loaded overpack itself. The HI-STAR 100 System is stored in either a vertical or horizontal orientation. A typical vertical-orientation ISFSI storage pattern is illustrated in Figure 1.4.1, which shows an array in a rectangular layout pattern. The required center-to-center spacing between the modules (layout pitch), guided by heat transfer considerations, is specified to be 12 feet in both orthogonal directions. The pitch may be increased to suit facility considerations.

A typical horizontal-orientation ISFSI storage pattern is illustrated in Figure 1.4.2. In horizontal storage, the design-basis lateral distance between cask centerlines is 18 feet (5.5 m) to provide practical access for handling equipment and the design-basis axial cask-to-cask clearance is 5 feet (1.5 m). Site-specific layouts for arrays of horizontal casks are evaluated pursuant to 10CFR72.212.

Vertically-oriented and horizontally-oriented casks will not be combined within an array, and a separation of at least 18 feet (5.5 m) will be maintained between adjacent groups of vertical and horizontal casks.

Table 1.4.1
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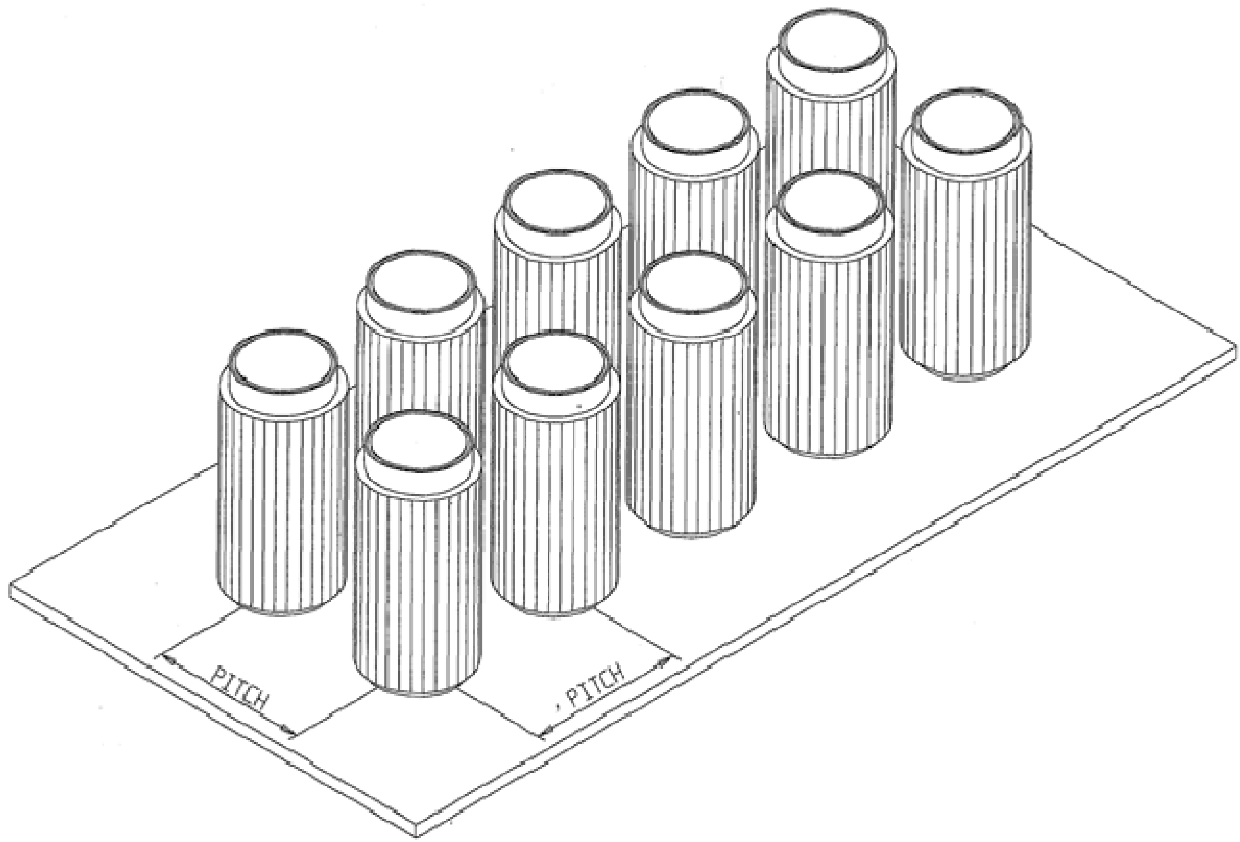

FIGURE 1.4.1; HI-STAR 100 TYPICAL ISFSI STORAGE PATTERN FOR VERTICAL STORAGE

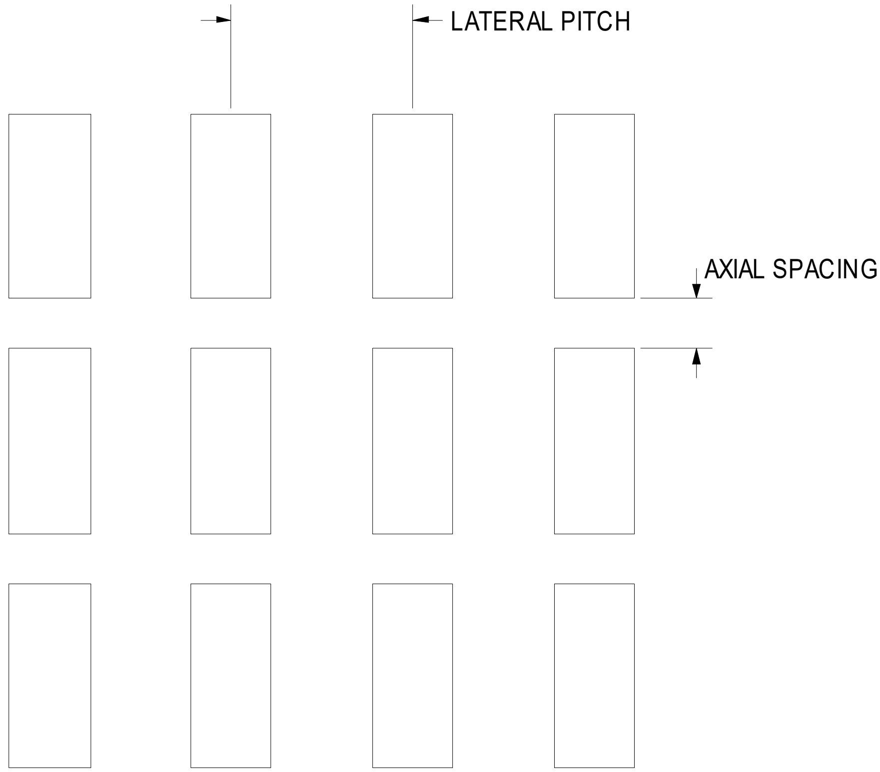

FIGURE 1.4.2; HI-STAR 100 TYPICAL ISFSI STORAGE PATTERN FOR HORIZONTAL STORAGE

1.5 GENERAL ARRANGEMENT DRAWINGS

The following HI-STAR 100 System drawings are provided in this section:

Drawing Number/Sheet	Description	Rev.
3913	HI-STAR 100 Overpack	10
3923	MPC Enclosure Vessel (Used with MPC-24 and 68/68F/68FF Fuel Basket)	10
3926	MPC-24 Fuel Basket Assembly	5
3928	MPC-68/68F Fuel Basket Assembly Note, this drawing also refers to the MPC-68FF, which is not part of this FSAR and not allowed for storage in the HI-STAR 100	5
3923	MPC Enclosure Vessel (Used with MPC-32 Fuel Basket)	25
3927	MPC-32 Fuel Basket Assembly	16

Changes authorized under ECOs 1023-88, 1021-149, 1022-107, and 1023-86 (72.48s # 1422 and 1407) have been evaluated for the HI-STAR 100 system and are considered acceptable. However, those changes have been incorporated into drawing revisions that include other changes not authorized for the HI-STAR 100 system, so the drawings are not updated in this FSAR.

PROPRIETARY DRAWINGS WITHHELD IN ACCORDANCE WITH 10 CFR 2.390

1.6 REGULATORY COMPLIANCE

Chapter 1 provides a general description of the HI-STAR 100 System which allows a reviewer to obtain a basic understanding of the system, its components, and the protection afforded for the health and safety of the public. The chapter has been written to provide the following pertinent information to allow verification of compliance with 10CFR72, NUREG-1536, and Regulatory Guide 3.61:

- A general description and discussion of the HI-STAR 100 System is presented in Sections 1.1 and 1.2 of the FSAR with special attention to design and operating characteristics, unusual or novel design features, and principal safety features.

- Drawings for structures, systems, and components (SSCs) important to safety are presented in Section 1.5 of the FSAR.

- Specifications for the spent fuel to be stored in the HI-STAR 100 System are provided in FSAR Subsection 1.2.3. Additional details concerning these specifications are provided in Section 2.1 of the FSAR.

- The technical qualifications of the Holtec International to engage in the proposed activities are identified in Section 1.3 of the FSAR.

- The quality assurance program and implementing procedures are described in Chapter 13 of the FSAR.

- The HI-STAR 100 System SAR was submitted, Docket No. 71-9261, and certified by the US NRC under 10CFR71 for use in transportation.

1.7 REFERENCES

[1.0.1] 10CFR Part 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste*.

[1.0.2] Regulatory Guide 3.61 (Task CE306-4) "Standard Format for a Topical Safety Analysis Report for a Spent Fuel Storage Cask", USNRC, February 1989.

[1.0.3] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", U.S. Nuclear Regulatory Commission, January 1997.

[1.0.4] 10CFR Part 71, "Packaging and Transportation of Radioactive Materials", Title 10 of the Code of Federal Regulations, 1998 Edition, Office of the Federal Register, Washington, D.C.

[1.0.5] HI-STAR 100 Transportation SER, Amendment 5, Docket 71-9261, dated October 12, 2006

[1.0.6] "Cladding Considerations for the Transportation and Storage of Spent Fuel," USNRC Interim Staff Guidance (ISG)-11, Revision 3, November 17, 2003

[1.0.7] HI-STORM FW Storage SER, Amendment 0, Docket 72-1032, dated July 14, 2011

[1.0.8] HI-STAR 100 Transportation SER, Amendment 2, Docket 71-9261, dated May 31, 2002

[1.0.9] HI-STAR 100 Transportation SER, Amendment 8, Docket 71-9261, dated October 12, 2010

[1.0.10] HI-STORM 100 Storage SER, Amendment 2, Docket 72-1014, dated June 7, 2005

[1.0.11] HI-STORM UMAX Storage SER, Amendment 0, Docket 72-1040, dated April 2, 2015

[1.0.12] HI-STORM 100 Storage SER, Amendment 1, Docket 72-1014, dated July 15, 2002

[1.0.13] HI-STAR 100 Transportation SER, Amendment 0, Docket 71-9261, March 31, 1999

[1.1.1] U.S. Department of Energy, "Multi-Purpose Canister (MPC) Subsystem Design Procurement Specification", Document No. DBG000000-01717-6300-00001, Rev. 5, January 11, 1996.

[1.1.2] U.S. Department of Energy, "MPC Transportation Cask Subsystem Design Procurement Specification", Document No. DBF 000000-01717-6300-00001, Rev. 5, January 11, 1996.

[1.2.1] U.S. NRC Information Notice 96-34, "Hydrogen Gas Ignition During Closure Welding of a VSC-24 Multi-Assembly Scale Basket".

[1.2.2] An Experimental Investigation into the Behavior of Boral for Dry Storage Applications, Holtec Report HI-2033067, Current Revision

[1.2.3] Qualification of METAMIC for Spent Fuel Storage Application, EPRI 1003137, Final Report, October 2001.

[1.2.4] Sourcebook For Metamic Performance Assessment, Holtec Report No. 2043215, Current Revision

[1.2.5] HI-STORM 100 Final Safety Analysis Report, Holtec Report No. HI-2002444, Current Revision

[1.2.6] HI-STAR 100 Safety Analysis Report, Holtec Report No. HI-951251, Current Revision.

[1.2.7] HI-STORM FW Final Safety Analysis Report, Holtec Report No. HI-2114830, Current Revision.

[1.2.8] HI-STAR 60 Safety Analysis Report, Holtec Report No. HI-2073710, Current Revision.

[1.2.9] HI-STORM UMAX Safety Analysis Report, Holtec Report No. HI-2115090, Current Revision

[1.3.1] 10CFR Part 50, Domestic Licensing of Production and Utilization Facilities

APPENDIX 1.A: ALLOY X DESCRIPTION

1.A ALLOY X DESCRIPTION

1.A.1 Alloy X Introduction

Alloy X is used within this licensing application to designate a group of stainless steel alloys. Alloy X can be any one of the following alloys:

- Type 316
- Type 316LN
- Type 304
- Type 304LN

Qualification of structures made of Alloy X is accomplished by using the least favorable mechanical and thermal properties of the entire group for all MPC mechanical, structural, neutronic, radiological, and thermal conditions. The Alloy X approach is conservative because no matter which material is ultimately utilized, the Alloy X approach guarantees that the performance of the MPC will meet or exceed the analytical predictions.

This appendix defines the least favorable material properties of Alloy X.

1.A.2 Alloy X Common Material Properties

Several material properties do not vary significantly from one Alloy X constituent to the next. These common material properties are as follows:

- density
- specific heat
- Young's Modulus (Modulus of Elasticity)
- Poisson's Ratio

The values utilized for this licensing application are provided in their appropriate chapters.

1.A.3 Alloy X Least Favorable Material Properties

The following material properties vary between the Alloy X constituents:

- Design Stress Intensity (S_m)
- Tensile (Ultimate) Strength (S_u)
- Yield Strength (S_y)
- Coefficient of Thermal Expansion (α)
- Coefficient of Thermal Conductivity (k)

Each of these material properties are provided in the ASME Code Section II [1.A.1]. Tables 1.A.1

through 1.A.5 provide the ASME Code values for each constituent of Alloy X along with the least favorable value utilized in this licensing application. The ASME Code only provides values to -20ºF (-29°C). The design temperature of the MPC is -40 ºF (-40°C) to 725 ºF (385°C) as stated in Table 1.2.2. Most of the above-mentioned properties become increasingly favorable as the temperature drops. Conservatively, the values at the lowest design temperature for the HI-STAR 100 System have been assumed to be equal to the lowest value stated in the ASME Code. The lone exception is the thermal conductivity. The thermal conductivity decreases with the decreasing temperature. The thermal conductivity value for -40 ºF (-40°C) is linearly extrapolated from the 70 ºF (21°C) value using the difference from 70 ºF (21°C) to 100 ºF (38°C).

The Alloy X material properties are the minimum values of the group for the design stress intensity, tensile strength, yield strength, and coefficient of thermal conductivity. Using minimum values of design stress intensity is conservative because lower design stress intensities lead to lower allowables that are based on design stress intensity. Similarly, using minimum values of tensile strength and yield strength is conservative because lower values of tensile strength and yield strength lead to lower allowables that are based on tensile strength and yield strength. When compared to calculated values, these lower allowables result in factors of safety that are conservative for any of the constituent materials of Alloy X. Further discussion of the justification for using the minimum values of coefficient of thermal conductivity is given in Chapter 4. The maximum and minimum values are used for the coefficient of thermal expansion of Alloy X. The maximum and minimum coefficients of thermal expansion are used as appropriate in this submittal.

1.A.4 References

[1.A.1] ASME Boiler & Pressure Vessel Code Section II, 1995 ed. with Addenda.

Table 1.A.1

ALLOY X AND CONSTITUENT DESIGN STRESS INTENSITY (S_m) vs. TEMPERATURE

Temp, °F (°C)	Type 304	Type 304LN	Type 316	Type 316LN	Alloy X (minimum of constituent values)
-40 (-40)	20.0 (138)	20.0 (138)	20.0 (138)	20.0 (138)	20.0 (138)
100 (38)	20.0 (138)	20.0 (138)	20.0 (138)	20.0 (138)	20.0 (138)
200 (93)	20.0 (138)	20.0 (138)	20.0 (138)	20.0 (138)	20.0 (138)
300 (149)	20.0 (138)	20.0 (138)	20.0 (138)	20.0 (138)	20.0 (138)
400 (204)	18.7 (129)	18.7 (129)	19.3 (133)	18.9 (130)	18.7 (129)
500 (260)	17.5 (121)	17.5 (121)	18.0 (124)	17.5 (121)	17.5 (121)
600 (316)	16.4 (113)	16.4 (113)	17.0 (117)	16.5 (114)	16.4 (113)
650 (343)	16.2 (112)	16.2 (112)	16.7 (115)	16.0 (110)	16.0 (110)
700 (371)	16.0 (110)	16.0 (110)	16.3 (112)	15.6 (108)	15.6 (108)
750 (399)	15.6 (108)	15.6 (108)	16.1 (111)	15.2 (105)	15.2 (105)
800 (427)	15.2 (105)	15.2 (105)	15.9 (110)	14.9 (103)	14.9 (103)

Notes:

1. Source: Table 2A on pages 314, 318, 326, and 330 of [1.A.1].
2. Units of design stress intensity values are given in ksi and (MPa) for reference.

Table 1.A.2

ALLOY X AND CONSTITUENT TENSILE STRENGTH (S_u) vs. TEMPERATURE

Temp °F (°C)	Type 304		Type 304LN		Type 316		Type 316LN		Alloy X (Note 4)	
	SA-240 (Note 3)	SA-336 (Note 3)	SA-240 (Note 3)	SA-336 (Note 3)	SA-240 (Note 3)	SA-336 (Note 3)	SA-240 (Note 3)	SA-336 (Note 3)	SA-240 (Note 3)	SA-336 (Note 3)
-40 (-40)	75.0 (517)	70.0 (483)	75.0 (517)	70.0 (483)	75.0 (517)	70.0 (483)	75.0 (517)	70.0 (483)	75.0 (517)	70.0 (483)
100 (38)	75.0 (517)	70.0 (483)	75.0 (517)	70.0 (483)	75.0 (517)	70.0 (483)	75.0 (517)	70.0 (483)	75.0 (517)	70.0 (483)
200 (93)	71.0 (490)	66.2 (456)	71.0 (490)	66.2 (456)	75.0 (517)	70.0 (483)	75.0 (517)	70.0 (483)	71.0 (490)	66.2 (456)
300 (149)	66.0 (455)	61.5 (424)	66.0 (455)	61.5 (424)	73.4 (506)	68.5 (472)	70.9 (489)	66.0 (455)	66.0 (455)	61.5 (424)
400 (204)	64.4 (444)	60.0 (414)	64.4 (444)	60.0 (455)	71.8 (495)	67.0 (462)	67.1 (463)	62.6 (432)	64.4 (444)	60.0 (414)
500 (260)	63.5 (438)	59.3 (409)	63.5 (438)	59.3 (409)	71.8 (495)	67.0 (462)	64.6 (445)	60.3 (416)	63.5 (438)	59.3 (409)
600 (316)	63.5 (438)	59.3 (409)	63.5 (438)	59.3 (409)	71.8 (495)	67.0 (462)	63.1 (435)	58.9 (406)	63.1 (435)	58.9 (406)
650 (343)	63.5 (438)	59.3 (4090	63.5 (438)	59.3 (409)	71.8 (495)	67.0 (462)	62.8 (433)	58.6 (404)	62.8 (433)	58.6 (404)
700 (371)	63.5 (438)	59.3 (409)	63.5 (438)	59.3 (409)	71.8 (495)	67.0 (462)	62.5 (431)	58.4 (403)	62.5 (431)	58.4 (403)
750 (399)	63.1 (435)	58.9 (406)	63.1 (435)	58.9 (406)	71.4 (492)	66.5 (459)	62.2 (429)	58.1 (401)	62.2 (429)	58.1 (401)
800 (427)	62.7 (432)	58.5 (403)	62.7 (432)	58.5 (403)	70.9 (489)	66.2 (456)	61.7 (425)	57.6 (397)	61.7 (425)	57.6 (397)

Notes:

1. Source: Table U on pages 437, 439, 441, and 443 of [1.A.1].

2. Units of tensile strength are given in ksi and (MPa) for reference.

3. The ultimate stress of Alloy X is dependent on the product form of the material (i.e., forging vs. plate). Values in the SA-336 column are based on SA-336 forged materials (type F304, F304LN, F316, and F316LN), which are used solely for the one-piece construction MPC lids. All other values correspond to SA-240 plate material.

4. Tensile strengths for Alloy X material are the minimum of constituent values.

Table 1.A.3

ALLOY X AND CONSTITUENT YIELD STRESSES (S_y) vs. TEMPERATURE

Temp. (°F)	Type 304	Type 304LN	Type 316	Type 316LN	Alloy X (minimum of constituent values)
-40 (-40)	30.0 (207)	30.0 (207)	30.0 (207)	30.0 (207)	30.0 (207)
100 (38)	30.0 (207)	30.0 (207)	30.0 (207)	30.0 (207)	30.0 (207)
200 (93)	25.0 (172)	25.0 (172)	25.8 (178)	25.5 (176)	25.0 (172)
300 (149)	22.5 (155)	22.5 (155)	23.3 (161)	22.9 (158)	22.5 (155)
400 (204)	20.7 (143)	20.7 (143)	21.4 (148)	21.0 (145)	20.7 (143)
500 (260)	19.4 (134)	19.4 (134)	19.9 (137)	19.4 (134)	19.4 (134)
600 (316)	18.2 (126)	18.2 (126)	18.8 (130)	18.3 (126)	18.2 (126)
650 (343)	17.9 (123)	17.9 (123)	18.5 (128)	17.8 (123)	17.8 (123)
700 (371)	17.7 (122)	17.7 (122)	18.1 (125)	17.3 (119)	17.3 (119)
750 (399)	17.3 (119)	17.3 (119)	17.8 (123)	16.9 (117)	16.9 (117)
800 (427)	16.8 (116)	16.8 (116)	17.6 (121)	16.6 (115)	16.6 (115)

Notes:

1. Source: Table Y-1 on pages 518, 519, 522, 523, 530, 531, 534, and 535 of [1.A.1].
2. Units of yield stress are given in ksi and (MPa) for reference.

Table 1.A.4

ALLOY X AND CONSTITUENT COEFFICIENT OF THERMAL EXPANSION
vs. TEMPERATURE

Temp, ºF (°C)	Type 304 and Type 304LN	Type 316 and Type 316LN	Alloy X Maximum	Alloy X Minimum
-40 (-40)	8.55 (15.4)	8.54 (15.4)	8.55 (15.4)	8.54 (15.4)
100 (38)	8.55 (15.4)	8.54 (15.4)	8.55 (15.4)	8.54 (15.4)
150 (66)	8.67 (15.6)	8.64 (15.6)	8.67 (15.6)	8.64 (15.6)
200 (93)	8.79 (15.8)	8.76 (15.8)	8.79 (15.8)	8.76 (15.8)
250 (121)	8.90 (16.0)	8.88 (16.0)	8.90 (16.0)	8.88 (16.0)
300 (149)	9.00 (16.2)	8.97 (16.1)	9.00 (16.2)	8.97 (16.1)
350 (177)	9.10 (16.4)	9.11 (16.4)	9.11 (16.4)	9.10 (16.4)
400 (204)	9.19 (16.5)	9.21 (16.6)	9.21 (16.6)	9.19 (16.5)
450 (232)	9.28 (16.7)	9.32 (16.8)	9.32 (16.8)	9.28 (16.7)
500 (260)	9.37 (16.9)	9.42 (17.0)	9.42 (17.0)	9.37 (16.9)
550 (288)	9.45 (17.0)	9.50 (17.1)	9.50 (17.1)	9.45 (17.0)
600 (316)	9.53 (17.2)	9.60 (17.3)	9.60 (17.3)	9.53 (17.2)
650 (343)	9.61 (17.3)	9.69 (17.4)	9.69 (17.4)	9.61 (17.3)
700 (371)	9.69 (17.4)	9.76 (17.6)	9.76 (17.6)	9.69 (17.4)
750 (399)	9.76 (17.6)	9.81 (17.7)	9.81 (17.7)	9.76 (17.6)
800 (427)	9.82 (17.7)	9.90 (17.8)	9.90 (17.8)	9.82 (17.7)

Notes:

1. Source: Table TE-1 on pages 590 and 591 of [1.A.1].
2. Units of coefficient of thermal expansion are given in./in.- ºF x 10^{-6} and (m/m- ºC x 10^{-6}) for reference.

Table 1.A.5
ALLOY X AND CONSTITUENT THERMAL CONDUCTIVITY vs. TEMPERATURE

Temp, ºF (°C)	Type 304 and Type 304LN	**Type 316** and Type 316LN	Alloy X (minimum of constituent values)
-40 (-40)	8.23 (14.24)	6.96 (12.05)	6.96 (12.05)
70 (21)	8.6 (14.9)	7.7 (13.3)	7.7 (13.3)
100 (38)	8.7 (15.1)	7.9 (13.7)	7.9 (13.7)
150 (66)	9.0 (15.58)	8.2 (14.2)	8.2 (14.2)
200 (93)	9.3 (16.1)	8.4 (14.5)	8.4 (14.5)
250 (121)	9.6 (16.6)	8.7 (15.1)	8.7 (15.1)
300 (149)	9.8 (17)	9.0 (15.6)	9.0 (15.6)
350 (177)	10.1 (17.48)	9.2 (15.9)	9.2 (15.9)
400 (204)	10.4 (18)	9.5 (16.4)	9.5 (16.4)
450 (232)	10.6 (18.35)	9.8 (17)	9.8 (17)
500 (260)	10.9 (18.87)	10.0 (17.31)	10.0 (17.31)
550 (288)	11.1 (19.21)	10.3 (17.83)	10.3 (17.83)
600 (316)	11.3 (19.56)	10.5 (18.17)	10.5 (18.17)
650 (343)	11.6 (20.08)	10.7 (18.52)	10.7 (18.52)
700 (371)	11.8 (20.42)	11.0 (19.04)	11.0 (19.04)
750 (399)	12.0 (20.77)	11.2 (19.28)	11.2 (19.28)
800 (427)	12.2 (21.11)	11.5 (19.9)	11.5 (19.9)

Notes:

1. Source: Table TCD on page 606 of [1.A.1].
2. Units of thermal conductivity are Btu/hr-ft-ºF and (W/mºC) for reference.

FIGURES 1.A.1 – 1.A.5 INTENTIONALLY DELETED

APPENDIX 1.B: HOLTITE-A ™ MATERIAL DATA
(Total of 4 Pages Including This Page)

The information provided in this appendix describes the neutron absorber material, Holtite-A for the purpose of confirming its suitability for use as a neutron shield material in spent fuel storage casks. Holtite-A is one of the family of Holtite neutron shield materials denoted by the generic name Holtite™. It is currently the only neutron shield material approved for installation in the HI-STAR 100 cask. It is chemically identical to NS-4-FR which was originally developed by Bisco Inc. and used for many years as a shield material with B_4C or Pb added.

Holtite-A contains aluminum hydroxide ($Al(OH)_3$) in an epoxy resin binder. Aluminum hydroxide is also known by the industrial trade name of aluminum tri-hydrate or ATH. ATH is often used commercially as a fire-retardant. Holtite-A contains approximately 62% ATH supported in a typical 2-part epoxy resin as a binder. Holtite-A contains 1% (nominal) by weight B_4C, a chemically inert material added to enhance the neutron absorption property. Pertinent properties of Holtite-A are listed in Table 1.B.1.

The essential properties of Holtite-A are:

1. the hydrogen density (needed to thermalize neutrons),

2. thermal stability of the hydrogen density, and

3. the uniformity in distribution of B_4C needed to absorb the thermalized neutrons.

ATH and the resin binder contain nearly the same hydrogen density so that the hydrogen density of the mixture is not sensitive to the proportion of ATH and resin in the Holtite-A mixture. B_4C is added as a finely divided powder and does not settle out during the resin curing process. Once the resin is cured (polymerized), the ATH and B_4C are physically retained in the hardened resin. Qualification testing for B_4C throughout a column of Holtite-A has confirmed that the B_4C is uniformly distributed with no evidence of settling or non-uniformity. Furthermore, an excess of B_4C is specified in the Holtite-A mixing and pouring procedure as a precaution to assure that the B_4C concentration is always adequate throughout the mixture.

The specific gravity specified in Table 1.B.1 does not include an allowance for weight loss. The specific gravity assumed in the shielding analysis includes a 4% reduction to conservatively account for potential weight loss at the design temperature of 300°F (148.9°C) or an inability to reach theoretical density. Tests on the stability of Holtite-A were performed by Holtec International. The results of the tests are summarized in Holtec Reports HI-2002396, “Holtite-A Development History and Thermal Performance Data” and HI-2002420, “Results of Pre- and Post-Irradiation Test Measurements.” The information provided in these reports demonstrates that Holtite-A™ possesses the necessary thermal and radiation stability characteristics to function as a reliable shielding material in the HI-STAR 100 overpack.

The Holtite-A is encapsulated in the HI-STAR 100 overpack and, therefore, should experience a very small weight reduction during the design life of the HI-STAR 100 System.

The data and test results confirm that Holtite-A remains stable under design thermal and radiation conditions, the material properties meet or exceed that assumed in the shielding analysis, and the B_4C remains uniformly distributed with no evidence of settling or non-uniformity.

Based on the information described above, Holtite-A meets all of the requirements for an acceptable neutron shield material.

Table 1.B.1

REFERENCE PROPERTIES OF HOLTITE-A NEUTRON SHIELD MATERIAL

PHYSICAL PROPERTIES	
% ATH	62 nominal
Specific Gravity	1.68 g/cc nominal
Max. Continuous Operating Temperature	300°F (148.9°C)
Hydrogen Density	0.096 g/cc minimum
Radiation Resistance	Excellent
CHEMICAL PROPERTIES (Nominal)	
wt% Aluminum	21.5
wt% Hydrogen	6.0
wt% Carbon	27.7
wt% Oxygen	42.8
wt% Nitrogen	2.0
wt% B_4C	1.0

PAGES 1.B-4 THROUGH 1.B-20 INTENTIONALLY DELETED

APPENDIX 1.C: MISCELLANEOUS MATERIAL DATA
(Total of 1 Pages Including This Page)

The information provided in this appendix specifies the thermal expansion foam (silicone sponge), paint, and anti-seize lubricant properties and demonstrates their suitability for use in spent nuclear fuel storage casks.

HT-870 silicone sponge or equivalent is specified as a thermal expansion foam to be placed in the overpack outer enclosure with the neutron shield. Due to differing thermal expansion of the neutron shield and outer enclosure carbon steel, the silicone sponge is provided to compress and allow the neutron shield material to expand.

Silicone has a long and proven history in the nuclear industry. Silicone is highly resistant to degradation as a result of radiation at the levels required for the HI-STAR 100 System. Silicone is inherently inert and stable and will not react with the metal surfaces or neutron shield material. Additionally, typical operating temperatures for silicone sponges range from -50°F (10°C) to 400°F (204.4°C).

Thermaline 450 or a functionally equivalent paint/coating is specified to coat the inner cavity of the overpack and Carboline 890 or a functionally equivalent paint/coating is specified to coat the external surfaces of the overpack. As can be seen from the product data sheets, the paints are suitable for the design temperatures (see Table 2.2.3) and chemical environment.

Nuclear grade anti-seize lubricant, N-5000, from FEL-PRO (or equivalent) is specified as the lubricant for the overpack closure bolts. The lubricant is formulated to have the lowest practical levels of halogens, sulfur, and heavy metals.

CHAPTER 2: PRINCIPAL DESIGN CRITERIA†

This chapter contains a compilation of design criteria applicable to the HI-STAR 100 System. The loadings and conditions prescribed herein for the MPC, particularly those pertaining to mechanical accidents, are far more severe in most cases than those required for 10CFR72 compliance. The underlying reason for the more stringent design criteria selected in this submittal is the dual-purpose nature (storage and transport) of the HI-STAR 100 System and its approved 10CFR71 certification [2.0.1]. This chapter sets forth the loading conditions and relevant acceptance criteria; it does not provide results of any analyses. The analyses and results carried out to demonstrate compliance with the design criteria are presented in the subsequent chapters of this report.

Much of the new information in this chapter is directly extracted or adopted from previously NRC approved Holtec dockets; this information is shown in *italics*. In Chapter 2, this information was extracted from References [1.2.5], [1.2.6] and [1.2.7]. All changes in this revision are marked with revision bars.

2.0 PRINCIPAL DESIGN CRITERIA

The design criteria for the MPC and HI-STAR overpack are summarized in Tables 2.0.1 and 2.0.2, respectively, and described in the sections that follow.

2.0.1 MPC Design Criteria

General

The Design Life of the MPCs is provided in Table 2.0.1, which meets the requirements of 10CFR72 [2.0.2]. The adequacy of the MPC design for the design life is discussed in Subsection 3.4.11.

Because the MPCs have also been previously certified in the HI-STORM 100 docket for storage, the design criteria in this section are aligned with those previously approved in the HI-STORM 100 docket [1.2.5]. The only exception to this is the permissible heat load (which is affected by the heat dissipation capability of the overpack and is set equal to the low value in the HI-STAR transport SAR [1.2.6]).

Structural

The MPC is classified as important to safety. The MPC structural components include the internal fuel basket and the enclosure vessel. The fuel basket is designed and fabricated as a core support structure, in accordance with the applicable requirements of Section III, Subsection NG of the ASME Code [2.0.3], with certain NRC-approved alternatives, as discussed in Subsection 2.2.4. The enclosure vessel is designed and fabricated as a Class 1 component pressure vessel in accordance with Section III, Subsection NB of the ASME Code, to the maximum extent practicable, as

† Wherever multiple units are shown, the US units are the governing value, and the SI units are shown for information only.

discussed in Subsection 2.2.4. The principal exceptions are the MPC lid, vent and drain cover plates, and closure ring welds to the MPC lid and shell, as discussed in Subsection 2.2.4. In addition, the threaded holes in the MPC lid are designed in accordance with the requirements of ANSI N14.6 [2.0.4] for critical lifts to facilitate vertical MPC transfer.

The MPC closure welds are partial penetration welds that are structurally qualified by analysis, as presented in Chapter 3. The MPC lid and closure ring welds are inspected by performing a liquid penetrant examination of the root pass and final weld surface, in accordance with the drawings contained in Section 1.5. The integrity of the MPC lid weld is further verified by performing a volumetric (or multi-layer liquid penetrant) examination, and a Code pressure test.

The structural analysis of the MPC, in conjunction with the redundant closures and nondestructive examination, hydrostatic pressure testing, and helium leak testing performed during MPC fabrication and MPC closure, provides assurance of canister closure integrity in lieu of the specific weld joint requirements of the ASME Code, Section III, Subsection NB.

Compliance with the ASME Code as it is applied to the design and fabrication of the MPC, and the associated justification, are discussed in Subsection 2.2.4. Compliance with the ASME Code is fully consistent with that used by other canister-based dry storage systems previously approved by the NRC.

The MPC is designed for all design basis normal, off-normal, and postulated accident conditions, as defined in Section 2.2. The load combinations for which the MPC is designed are defined in Subsection 2.2.7. In addition, the maximum allowable weight and dimensions of a fuel assembly to be stored in the MPC are limited in accordance with Subsection 2.1.4.

Thermal

The thermal design and operation of the MPC in the HI-STAR 100 cask meets the intent of the review guidance contained in ISG-11, Revision 3 [2.0.5]. Specifically, the ISG-11 provisions that are explicitly invoked and satisfied are:

i. The thermal acceptance criteria for all commercial spent fuel (CSF) authorized by the USNRC for operation in a commercial reactor are unified into one set of requirements.

ii. The maximum value of the calculated temperature for all CSF under long-term normal conditions of storage must remain below 400ºC (752ºF). For short-term operations, including canister drying, helium backfill, and on-site cask transport operations, the fuel cladding temperature must not exceed 400ºC (752ºF) for high burnup fuel (HBF) and 570ºC (1058ºF) for moderate burnup fuel. (Note, high burnup fuel is not included in the contents for the HI-STAR 100 storage system, see Tables 2.1.13, 2.1.14, and 2.1.15, and is therefore not allowed for storage in the HI-STAR 100.)

iii. The maximum fuel cladding temperature as a result of an off-normal or accident event must not exceed 570ºC (1058ºF).

iv. During loading operations, repeated thermal cycling may occur but are limited to cladding temperature variations that are less than 65ºC (117ºF) each and the number of excursions to less than 10.

To achieve compliance with the above criteria, certain design and operational changes are necessary, as summarized below.

i. The peak fuel cladding temperature limit (PCT) for long term storage operations and short term operations is generally set at 400ºC (752ºF). However, for MPCs containing <u>all</u> moderate burnup fuel, the fuel cladding temperature limit for short-term operations is set at 570ºC (1058ºF) because the nominal fuel cladding stress is shown to be less than 90 MPa [2.0.6]. Appropriate analyses have been performed as discussed in Chapter 4 and operating restrictions have been added to ensure these limits are met.

ii. A method of drying, such as forced helium dehydration (FHD) is used if the above temperature limits for short-term operations cannot be met.‡

iii. The off-normal and accident condition PCT limit remains unchanged at 570 ºC (1058ºF).

The MPC cavity is dried, either with FHD or vacuum drying, and then it is backfilled with high purity helium to promote heat transfer and prevent cladding degradation.

The design temperatures for the structural steel components of the MPC are based on the temperature limits provided in ASME Section II, Part D, tables referenced in ASME Section III, Subsection NB and NG, for those load conditions under which material properties are relied on for a structural load combination. The specific design temperatures for the components of the MPC are provided in Table 2.2.3.

The MPCs are designed for a bounding thermal source term, as described in Subsection 2.1.5. The maximum allowable heat load in the MPC is limited in accordance with the Section 2.1.

Because the HI-STAR 100 overpack is also envisaged to serve as a transport package, the thermal analyses for all canisters in this docket have been performed with the thermosiphon effect neglected. The thermosiphon effect, as noted in the HI-STORM 100 and HI-STORM FW FSARs, is a major dissipation of heat in Holtec MPCs. However, the thermosiphon action is substantially reduced when the canister is horizontal. Therefore, for conservatism, the thermosiphon effect is not simulated in the thermal model for either vertical or horizontal storage.

<u>Shielding</u>

The allowable doses for an ISFSI using the HI-STAR 100 System are delineated in 10CFR72.104 and 72.106. Compliance with this criteria is necessarily site-specific and is to be demonstrated by the licensee, as discussed in Chapters 5 and 10.

‡ Note: In this FSAR vacuum drying is always accepted. This statement is for future design changes, when applicable.

The MPC provides axial gamma shielding at the top and bottom ends to maintain occupational exposures ALARA during canister closure and handling operations. The maximum allowable top axial dose rates for the MPC are controlled in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 10).

The MPCs are designed for the design basis fuel at the maximum burnup and minimum cooling times, as described in Subsection 2.1.6 and 5.2. The radiological source term for the MPCs are limited based on the burnup and cooling times specified in the tables in Section 2.1. Calculated dose rates for each MPC are provided in Section 5.1. These dose rates are used to perform an occupational exposure evaluation in accordance with 10CFR20, as discussed in Chapter 10.

Criticality

The MPCs provide criticality control for all design basis normal, off-normal, and postulated accident conditions, as discussed in Section 6.1. The effective neutron multiplication factor is limited to $k_{eff} < 0.95$ for fresh unirradiated intact and damaged fuel assemblies with optimum unborated water moderation and close reflection, including all biases, uncertainties, and MPC manufacturing tolerances.

Criticality control is maintained by the geometric spacing of the fuel assemblies and fixed borated neutron absorbing materials incorporated into the fuel basket assembly. The minimum specified boron concentration verified during Boral manufacture is further reduced by 25% for criticality analysis.

The guaranteed B-10 value in the Metamic, assured by the manufacturing process, is reduced by 10% (90% credit is taken for the Metamic) to accord with NUREG/CR-5661. No credit is taken for fuel burnup or integral poisons such as gadolinia in BWR fuel. In addition, burnup credit is not taken in PWR fuel. The soluble boron concentration requirements (for PWR fuel in MPC-24 and MPC-32) based on the initial enrichment of the fuel assemblies are delineated in Section 2.1 consistent with the criticality analysis described in Chapter 6.

The maximum allowable initial enrichment for fuel assemblies to be stored in each MPC are limited in accordance with the Section 2.1.

Confinement

The MPC provides for confinement of all radioactive materials for all design basis normal, off-normal, and postulated accident conditions. As discussed in Section 7.1, the HI-STAR 100 MPC design meets the guidance in Interim Staff Guidance (ISG)-18 [2.0.8] so that leakage of radiological matter from the confinement boundary is non-credible. Therefore, no confinement dose analysis is required or performed. The confinement function of the MPC is verified through pressure testing, helium leak testing of the MPC shell, base plate, and lid material along with the shell to base plate and shell to shell seam welds, and a rigorous weld examination regimen executed in accordance with the acceptance test program in Chapter 9.

Operations

There are no radioactive effluents that result from storage or transfer operations. Effluents generated during MPC loading and closure operations are handled by the plant's radwaste system and procedures under the licensee's 10CFR50 license.

Generic operating procedures for the HI-STAR 100 System are provided in Chapter 8. Detailed operating procedures are required to be developed by the licensee based on site-specific requirements that comply with the 10CFR50 Technical Specifications for the plant and the 10CFR72 Technical Specifications for the HI-STAR 100 System CoC.

Acceptance Tests and Maintenance

The fabrication acceptance basis and maintenance program to be applied to the MPCs are described in Chapter 9. The operational controls and limits to be applied to the MPCs are contained in Chapter 12 Technical Specifications. Application of these requirements will assure that the MPC is fabricated, operated, and maintained in a manner that satisfies the design criteria defined in this chapter.

Decommissioning

The MPCs are designed to be transportable in the HI-STAR 100 overpack and are not required to be unloaded prior to shipment off-site. Decommissioning of the HI-STAR 100 System is addressed in Section 2.4.

2.0.2 HI-STAR Overpack

General

The HI-STAR overpack design life is provided in Table 2.0.2, which satisfies the requirements of 10CFR72. The adequacy of the overpack design for the design life is discussed in Subsection 3.4.10.

Structural

The HI-STAR overpack is classified as important to safety. The HI-STAR overpack top flange, closure plate, inner shell, and bottom plate are designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NB, with certain NRC-approved alternatives (see Subsection 2.2.4). The remainder of the HI-STAR overpack steel structure is designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NF, to the maximum extent practical (see Subsection 2.2.4). Compliance with the ASME Code is fully consistent with that used by other dry storage systems previously approved by the NRC.

The overpack is designed for all normal, off-normal, and design basis accident condition loadings, as defined in Section 2.2. The physical characteristics of the MPCs for which the overpack is designed are defined in Chapter 1.

Thermal

The allowable temperatures for the structural steel components are based on the maximum temperature for which material properties and allowable stresses are provided in Section II of the ASME Code. The specific allowable temperatures for the structural steel components of the overpack are provided in Table 2.2.3. The allowable temperature for the Holtite-A neutron shield material specified in Table 2.2.3 is based on the data provided in Appendix 1.B.

The overpack is designed for extreme cold conditions, as discussed in Paragraph 2.2.2.2. The structural steel materials used for the overpack that are susceptible to brittle fracture are discussed in Section 3.1.

The overpack is designed for the maximum allowable heat load for steady-state normal conditions, in accordance with Section 2.1. The thermal characteristics of the MPCs for which the overpack is designed are defined in Chapter 4.

<u>Shielding</u>

The off-site dose for normal operating conditions and anticipated occurrences at the controlled area boundary to a real individual is limited by 10CFR72.104(a) to a maximum of 25 mrem/year (0.25 mSv/year) whole body, 75 mrem/year (0.75 mSv/year) thyroid, and 25 mrem/year (0.25 mSv/year) for other critical organs, including contributions from all nuclear fuel cycle operations. Since these limits are dependent on plant operations as well as site-specific conditions (e.g., the ISFSI design and proximity to the controlled area boundary, and the number and arrangement of loaded storage casks), the determination and comparison of ISFSI doses to this limit are necessarily site-specific. Dose rates for a typical ISFSI using the HI-STAR 100 System are provided in Chapters 5 and 10. The determination of site-specific ISFSI dose rates at the controlled area boundary and demonstration of compliance with regulatory limits shall be performed by the licensee in accordance with 10CFR72.212.

The overpack is designed to limit the calculated surface dose rate at the cask midplane for all MPCs to the design objective, as defined in Section 2.3. The overpack is also designed to maintain occupational exposures ALARA during MPC transfer operations, in accordance with 10CFR20. The calculated overpack dose rates are presented in Section 5.1. These dose rates are used to perform a generic occupational exposure estimate for MPC loading operations and a dose assessment for a typical ISFSI, as described in Chapter 10.

<u>Confinement</u>

The overpack is not defined as the confinement boundary for radioactive materials. Confinement during storage is provided by the MPC which is addressed in Chapter 7. The overpack provides physical protection and biological shielding for the MPC confinement boundary during MPC dry storage operations.

<u>Operations</u>

There are no radioactive effluents that result from MPC transfer or storage operations with the overpack. Effluents generated during MPC loading and closure operations are handled by the plant's radwaste system and procedures under the licensee's 10CFR50 license.

Generic operating procedures for the HI-STAR 100 System are provided in Chapter 8. The licensee is required to develop detailed operating procedures based on site-specific conditions and requirements that also comply with the applicable 10CFR50 Technical Specification requirements for the site and the HI-STAR 100 System Technical Specifications.

Acceptance Tests and Maintenance

The fabrication acceptance basis and maintenance program to be applied to the overpack are described in Chapter 9. The operational controls and limits to be applied to the overpack are contained in Chapter 12. Application of these requirements will assure that the overpack is fabricated, operated, and maintained in a manner that satisfies the design criteria defined in this chapter.

Decommissioning

Decommissioning considerations for the HI-STAR 100 System, including the overpack, are addressed in Section 2.4.

Table 2.0.1

MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Life:			
Design	50 yrs. (Table 1.2.2)	-	Table 1.2.2
Regulatory	20 yrs.	10CFR72.42(a) and 10CFR72.236(g)	-
Structural:			
Design Codes:			
Enclosure Vessel	ASME Code, Section III, Subsection NB	10CFR72.24(c)(4)	Subsection 2.0.1
Fuel Basket	ASME Code, Section III, Subsection NG	10CFR72.24(c)(4)	Subsection 2.0.1
MPC Lifting Points	ANSI N14.6/NUREG-0612	10CFR72.24(c)(4)	Paragraph 1.2.1.4
Design Dead Weights:			
Max. Loaded Canister (dry)	See Table 3.2.1	ANSI/ANS 57.9	Table 3.2.1
Empty Canister (dry)	See Table 3.2.1	ANSI/ANS 57.9	Table 3.2.1
Design Cavity Pressures:			
Normal:	100 psig (700 kPa)	ANSI/ANS 57.9	Paragraph 2.2.1.3

Table 2.0.1 (continued)

MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Off-Normal:	100 psig (700 kPa)	ANSI/ANS 57.9	Paragraph 2.2.2.1
Accident (Internal)	200 psig (1380 kPa)	ANSI/ANS 57.9	Paragraph 2.2.3.8
Accident (External)	60 psig (400 kPa)	ANSI/ANS 57.9	Paragraphs 2.2.3.6 and 2.2.3.10
Response and Degradation Limits	SNF assemblies confined in dry, inert environment	10CFR72.122(h)(l)	Subsection 2.0.1
Thermal:			
Maximum Design Temperatures:			
Structural Materials:			
Stainless Steel (Normal)	Table 2.2.3	ASME Code Section II, Part D	Table 2.2.3
Stainless Steel (Accident)	Table 2.2.3	ASME Code Section II, Part D	Table 2.2.3
Neutron Poison:			
Boral or METAMIC (normal)	Table 2.2.3	See Section 4.3.1	Table 2.2.3
Boral or METAMIC (accident)	Table 2.2.3	See Section 4.3.1	Table 2.2.3
Fuel Cladding:	Table 2.2.3	ISG-11 Rev 3	Table 2.2.3
Canister Backfill Gas	Helium	-	Chapter 12
Canister Backfill Pressure	Varies by MPC	-	Chapter 12
Short-Term Allowable Fuel Cladding Temperature	Table 2.2.3	*ISG-11 Rev 3*	Subsection 2.0.1 & section 4.3
Insolation	Protected by Overpack	10CFR71.71	-
Confinement:		10CFR72.128(a)(3) and	

Table 2.0.1 (continued)

MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
		10CFR72.236(d) and (e)	
Closure Welds:			
Shell Seams and Shell-to-Baseplate	Full Penetration	-	Section 1.5 and Table 9.1.3
MPC Lid	Multi-pass Partial Penetration	10CFR72.236(e)	Section 1.5 and Table 9.1.3
MPC Closure Ring	Partial Penetration		
Port Covers	Partial Penetration		
NDE:			
Shell Seams and Shell-to-Baseplate	100% RT or UT	NUREG-1536	Chapter 8 and Table 9.1.3
MPC Lid	Root Pass/Final Surface 100% PT and Volumetric or Multi-Layer PT	NUREG-1536	Chapter 8 and Table 9.1.3
Closure Ring	Root Pass/Final Surface 100% PT	NUREG-1536	Chapter 8 and Table 9.1.3
Port Covers	Root Pass/Final Surface 100% PT	NUREG-1536	Chapter 8 and Table 9.1.3
Leak Testing:			

Table 2.0.1 (continued)

MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Welds Tested	Shell seams, shell-to-baseplate, and port covers-to-MPC lid	-	Section 7.1, Chapters 8, and 9
Medium	Helium	*ANSI N14.5*	*Section 9.1*
Max. Leak Rate	*Leaktight*	*ANSI N14.5*	*Section 9.1*
Monitoring System	None	10CFR72.128(a)(1)	Paragraph 2.3.2.1
Pressure Testing:			
Minimum Test Pressure	*125 psig (860 kPa) (hydrostatic)* *120 psig (825 kPa) (pneumatic)*	-	Chapters 8 and 9
Welds Tested	MPC Lid-to-Shell, MPC shell seams, MPC shell-to-baseplate (Table 9.1.1)	-	Section 8.1 and Table 9.1.1
Medium	*Water or helium*	-	Section 8.1 and Chapter 9
Retrievability:			
Normal and Off-normal: Post (design basis) Accident	No Encroachment on Fuel Assemblies or Exceeding Fuel Assembly Deceleration Limits	10CFR72.122(f),(h)(1), & (l)	Sections 3.4, 3.5, and Subsection 3.1.2
Criticality:		10CFR72.124 & 10CFR72.236(c)	

Table 2.0.1 (continued)

MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Method of Control	*Fixed Borated Neutron Absorber, Geometry and Soluble Boron*	-	Subsection 2.3.4
Min. Boron Loading	See Licensing Drawing (Section 1.5)	-	Subsection 2.1.7
Max. k_{eff}	0.95	-	Sections 6.1 and Subsection 2.3.4
Min. Burnup	0.0 GWd/MTU (fresh fuel)	-	Section 6.1
Min. Soluble Boron Concentration	*Table 2.1.18 and 2.1.19 and Section 6.1*	-	*Table 2.1.18 and 2.1.19 and Section 6.1*
Radiation Protection/Shielding:		10CFR72.126, & 10CFR72.128(a)(2)	
MPC: (normal/off-normal/accident)			
MPC Closure	ALARA	10CFR20	Sections 10.1, 10.2, & 10.3
MPC Transfer	ALARA	10CFR20	Sections 10.1, 10.2, & 10.3
Exterior of Shielding: (normal/off-normal/accident)			
Storage Mode Position	See Table 2.0.2	10CFR20	Subsection 5.1.1
ISFSI Controlled Area Boundary	See Table 2.0.2	10CFR72.104 & 10CFR72.106	Subsection 5.1.1 and Chapter 10
Design Bases:		10CFR72.236(a)	
Spent Fuel Specification:			
Assemblies/Canister	Table 1.1.1	-	Table 1.2.1
Type of Cladding	*Stainless or Zircaloy* *	-	Table 2.1.6
Fuel Condition	*Intact, Damaged, or Debris*	-	Subsection 2.1.2 and 2.1.3

Table 2.0.1 (continued)

MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
	(Table 2.1.6 and 2.1.7)		& Table 2.1.6 and 2.1.7
* Also designed to accommodate failed fuel, stainless clad fuel, and MOX fuel (Tables 2.1.7 and 2.1.11 and Appendix B to CoC 72-1008)			
PWR Fuel Assemblies:			
Type/Configuration	Various	-	Table 2.1.3
Max. Burnup	*42,500 MWD/MTU (MPC-24)* *44,500 MWD/MTU (MPC-32)*	-	Table 2.1.13 Table 2.1.15
Max. Enrichment	Varies by fuel design	-	Table 2.1.3
Design Basis MPC Heat Load:			
MPC-24 Vertical *MPC*-24 Horizontal	19 kW 20 kW	-	
MPC-32 Vertical MPC-32 Horizontal	18.5 kW 20 kW	-	
Minimum Cooling Time:	5 years (MPC-24) 5 years (MPC-32)	-	Table 2.1.13 Table 2.1.15
Max. Weight of Contents (Dry) MPC-68, -68F, -24, -32:	*Table 2.1.6*	-	Table 2.1.6
Max. Fuel Assembly Length: (unirradiated nominal)	176.8 in. (4490 mm)	-	Table 2.1.6
Max. Fuel Assembly Width (unirradiated nominal)	8.54 in. (217 mm)	-	Table 2.1.6
Fuel Rod Fill Gas:			
Pressure (max.)	500 psig (2450 kPa)	-	Section 4.3 & Table 4.3.2
BWR Fuel Assemblies:			
Type	Various	-	Table 2.1.4
Max. Burnup	*37,600 MWd/MTU*	-	Table 2.1.14
Max. Enrichment	Varies by fuel design	-	Section 6.1

Table 2.0.1 (continued)

MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Basis MPC Heat Load:			
MPC-68, -68F	*18.5 kW*		
Minimum Cooling Time:	5 years (MPC-68)	-	Table 2.1.14
Max. Fuel Assembly Weight:			
w/channels	700 lb. (310 kg)	-	Table 2.1.6
Max. Fuel Assembly Length (unirradiated nominal)	176.2 in. (4475 mm)	-	Table 2.1.6
Max. Fuel Assembly Width (unirradiated nominal)	5.85 in. (150 mm)	-	Table 2.1.6
Fuel Rod Fill Gas:			
Pressure (max.)	147 psig (1000 kPa)	-	Table 4.3.5
Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Temperatures	See Table 2.0.2	ANSI/ANS 57.9	Paragraph 2.2.1.4
Handling:			Paragraph 2.2.1.2
Handling Loads	115% of Dead Weight	CMAA #70	Paragraph 2.2.1.2
Special Lifting Device Acceptance Criteria	1/10 Ultimate 1/6 Yield	ANSI N14.6	Paragraph 2.2.1.2
Attachment/Component Interface Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Paragraph 2.2.1.2
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Paragraph 2.2.1.2
Wet/Dry Loading	Wet or Dry	-	Paragraph 1.2.2.2
Transfer Orientation	Vertical or Horizontal	-	Paragraph 1.2.2.2
Storage Orientation	Vertical, *horizontal, or inclined*	-	Paragraph 1.2.2.2
Fuel Rod Rupture Releases:			

Table 2.0.1 (continued)

MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Fuel Rod Failures	1%	NUREG-1536	Paragraph 2.2.1.3
Fill Gases	100%	NUREG-1536	Paragraph 2.2.1.3
Fission Gases	30%	NUREG-1536	Paragraph 2.2.1.3
Snow and Ice	Protected by Overpack	ASCE 7-88	Paragraph 2.2.1.6
Off-Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Temperature	See Table 2.0.2	ANSI/ANS 57.9	Paragraph 2.2.2.2
Fuel Rod Rupture Releases:			
Fuel Rod Failures	10%	NUREG-1536	Paragraph 2.2.2.1
Fill Gases	100%	NUREG-1536	Paragraph 2.2.2.1
Fission Gases	30%	NUREG-1536	Paragraph 2.2.2.1
Design-Basis (Postulated) Accident Design Events and Conditions:		10CFR72.24(d)(2) & 10CFR72.94	
Tip Over	See Table 2.0.2	-	Paragraph 2.2.3.2
End Drop	See Table 2.0.2	-	Paragraph 2.2.3.1
Side Drop	See Table 2.0.2	-	Paragraph 2.2.3.1
Fire	See Table 2.0.2	10CFR72.122(c)	Paragraph 2.2.3.3
Fuel Rod Rupture Releases:			
Fuel Rod Failures	100%	NUREG-1536	Section 2.2
Fill Gases	100%	NUREG-1536	Section 2.2
Fission Gases	30%	NUREG-1536	Section 2.2
Particulates & Volatiles	See Table 7.3.1	-	Sections 2.2 and 7.3
Confinement Boundary Leakage	*Leaktight (helium)*	ANSI N14.5	Sections 2.2 and 7.3
Explosive Overpressure	Protected by Overpack	10CFR72.122(c)	Paragraph 2.2.3.10
Design Basis Natural Phenomenon Design Events and Conditions:		10CFR72.92 & 10CFR72.122(b)(2)	
Flood Water Depth	125 ft.(35 m)	ANSI/ANS 57.9	Paragraph 2.2.3.6
Seismic	See Table 2.0.2	10CFR72.102(f)	Paragraph 2.2.3.7

Table 2.0.1 (continued)

MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Wind	Protected by Overpack	ASCE-7-88	Paragraph 2.2.3.5
Tornado & Missiles	Protected by Overpack	RG 1.76 & NUREG-0800	Paragraph 2.2.3.5
Burial Under Debris	Adiabatic Heat-Up	-	Paragraph 2.2.3.12
Lightning	See Table 2.0.2	NFPA 78	Paragraph 2.2.3.11
Partial Blockage of MPC Basket Vent Holes	Crud Depth (Table 2.2.8)	ESEERCO Project EP91-29	Paragraph 2.2.3.4
Extreme Environmental Temp.	See Table 2.0.2	-	Paragraph 2.2.3.13

Table 2.0.2

OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Life:			
Design	50 yrs.	-	Subsection 2.0.2
Regulatory	20 yrs.	10CFR72.42(a) & 10CFR72.236(g)	
Structural:			
Design Codes:			
Inner Shell, Closure Plate, Top Flange, Bottom Plate, and Closure Plate Bolts			
Design	ASME Code Section III, Subsection NB	10CFR72.24(c)(4)	Subsection 2.0.2
Fabrication	ASME Code Section III, Subsection NB	10CFR72.24(c)(4)	Subsection 2.0.2
Remainder of Structural Steel			
Design	ASME Code Section III, Subsection NF	10CFR72.24(c)(4)	Subsection 2.0.2
Fabrication	ASME Code Section III, Subsection NF	10CFR72.24(c)(4)	Subsection 2.0.2
Design Weights:			
Max. Loaded MPC (Dry)	Table 3.2.1	ANSI/ANS 57.9	Table 3.2.1
Max. Empty Overpack:			
Assembled with Closure Plate	Table 3.2.1	ANSI/ANS 57.9	Table 3.2.1
Max. MPC/Overpack:	Table 3.2.1	ANSI/ANS 57.9	Table 3.2.1

Table 2.0.2 (continued)

OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Cavity Pressures	40 psig (275 kPa)(Normal) 40 psig (275 kPa) (Off-Normal) 60 psig (410 kPa) (Accident)	-	Table 2.2.1
Response and Degradation Limits	See Table 2.0.1	10CFR72.122(h)(1)	Subsection 2.0.1
External Design Pressure (Normal and off-normal condition)	1 atmosphere (101 kPa)	-	Paragraph 2.2.1.3
Thermal:			
Maximum Design Temperatures:			
Inner Shell (SA203-E *or SA350-LF3)*			
Normal Condition Maximum	Table 2.2.3	ASME Code Section II, Part D	Table 2.2.3
Off-Normal/Accident Condition Maximum	Table 2.2.3	ASME Code Section II, Part D	Table 2.2.3
Top Flange & Closure Plate (SA350-LF3)			
Normal Condition Maximum	Table 2.2.3	ASME Code Section II, Part D	Table 2.2.3
Off-Normal/Accident Condition Maximum	Table 2.2.3	ASME Code Section II, Part D	Table 2.2.3
Bottom Plate (SA350-LF3)			
Normal Condition Maximum	Table 2.2.3	ASME Code Section II, Part D	Table 2.2.3
Off-Normal/Accident	Table 2.2.3	ASME Code	Table 2.2.3

Table 2.0.2 (continued)

OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Condition Maximum		Section II, Part D	
Remainder of Steel Structure	Table 2.2.3	ASME Code Section II, Part D	Table 2.2.3
Neutron Shield	Table 2.2.3	Manufacturer's Test Data	Table 2.2.3
Insolation:	Averaged Over 24 Hours	10CFR71.71	4.4.1.1.8
Confinement§:		10CFR72.128(a)(3) & 10CFR72.236(d) & (e)	
Pressure Testing:			
Minimum Test Pressure	125 psig (860 kPa) (hydrostatic) 150 psig (1030 kPa) (pneumatic)	*ASME Code Section III, NB 10CFR71.85(b)*	Chapters 8 and 9
Max Leak Rate:	*4.3 x 10^{-6} atm cm^3/s (4.3 x 10^{-7} Pa m^3/s) (Helium)*	-	*Chapter 4 of [1.2.6]*
Min Test Sensitivity:	*2.15 x 10^{-6} std cm^3/s (2.15 x 10^{-7} Pa m^3/s) (Helium)*	-	*Chapter 4 of [1.2.6]*
Retrievability:			
Normal and Off-normal	No damage which precludes Retrieval of MPC or Exceeding Fuel Assembly Deceleration Limits	10CFR72.122(f),(h)(1), & (l)	Sections 3.5 and 3.4
Accident			Sections 3.5 and 3.4
Criticality:	Protection of MPC and Fuel Assemblies	10CFR72.124 & 10CFR72.236(c)	Section 6.1

§ Note: Overpack is not part of confinement boundary, but sealed to maintain helium and prepare for transport as a containment boundary.

Table 2.0.2 (continued)

OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Radiation Protection/Shielding:		10CFR72.126 & 10CFR72.128(a)(2)	
Overpack (Normal/Off-normal/Accident)			
Surface	ALARA	10CFR20	Chapters 5 and 10
Position	ALARA	10CFR20	Chapters 5 and 10
Beyond Controlled Area During Normal Operation and Anticipated Occurrences	25 mrem/yr. (0.25 mSv/yr) to whole body 75 mrem/yr. (0.75 mSv/yr) to thyroid 25 mrem/yr. (0.25 mSv/yr) to any organ	10CFR72.104	Sections 5.1, 7.2, and 10.4
On Controlled Area Boundary from Design Basis Accident	5 rem (0.05 Sv) to whole body or to any organ	10CFR72.106	Sections 5.1, 7.3, and 10.4
Design Bases:			
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1
Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Outside Temperatures:			
Annual Average	80 ºF (27 ºC)	ANSI/ANS 57.9	Paragraph 2.2.1.4
Handling:			
Handling Loads	115% of Dead Weight	CMAA #70	Paragraph 2.2.1.2
Special Lifting Device Acceptance Criteria	1/10 Ultimate 1/6 Yield	ANSI N14.6	Paragraph 2.2.1.2
Snow and Ice Load	100 lb./ft^2 (488 kg/m^2)	ASCE 7-88	Paragraph 2.2.1.6
Wet/Dry Loading	Wet/Dry	-	Paragraph 1.2.2.2
Storage Orientation	Vertical, *horizontal, or inclined*	-	Paragraph 1.2.2.2
Off-Normal Design Event		10CFR72.122(b)(1)	

Table 2.0.2 (continued)

OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Conditions:			
Ambient Temperature			
Minimum	-40 ºF (-40 ºC)	ANSI/ANS 57.9	Paragraph 2.2.2.2
Maximum	100 ºF (38 ºC)	ANSI/ANS 57.9	Paragraph 2.2.2.2
Design-Basis (Postulated) Accident Design Events and Conditions:		10CFR72.94	
Drop Cases:			
End	21 in. (530 mm)	-	Paragraph 2.2.3.1
Side	72 in. (1830 mm)	-	Paragraph 2.2.3.1
Tip-Over	Assumed (Non-mechanistic)	-	Paragraph 2.2.3.2
Fire:			
Duration	305 seconds	10CFR72.122(c)	Paragraph 2.2.3.3
Temperature	1,475 ºF (800 ºC)	10CFR72.122(c)	Paragraph 2.2.3.3
Fuel Rod Rupture	See Table 2.0.1	-	Paragraph 2.2.3.8
Flood			
Height	656 ft. (200 m)	RG 1.59	Paragraph 2.2.3.6
Velocity	13 ft/sec. (4 m/s)	RG 1.59	Paragraph 2.2.3.6
Seismic for vertical storage			
Max. ZPA Horizontal Ground (Max. ZPA Vertical Ground)	0.314g (w/1.0 vertical) 0.332g (w/0.75 vertical) 0.339g (w/0.667 vertical) 0.354g (w/0.5 vertical)	10CFR72.102(f)	Paragraph 2.2.3.7
Tornado			
Wind			
Max. Wind Speed	360 mph (580 km/h)	RG 1.76	Paragraph 2.2.3.5
Pressure Drop	3.0 psi (20 kPa)	RG 1.76	Paragraph 2.2.3.5
Missiles			Paragraph 2.2.3.5

Table 2.0.2 (continued)

OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Automobile			
Weight	Table 2.2.5	NUREG-0800	Table 2.2.5
Velocity	Table 2.2.5	NUREG-0800	Table 2.2.5
Rigid Solid Steel Cylinder (Artillery Shell)			
Weight	Table 2.2.5	NUREG-0800	Table 2.2.5
Velocity	Table 2.2.5	NUREG-0800	Table 2.2.5
Diameter	Table 2.2.5	NUREG-0800	Table 2.2.5
Steel Sphere			
Weight	Table 2.2.5	NUREG-0800	Table 2.2.5
Velocity	Table 2.2.5	NUREG-0800	Table 2.2.5
Diameter	Table 2.2.5	NUREG-0800	Table 2.2.5
Burial Under Debris	Adiabatic Heat-Up	-	Paragraph 2.2.3.12
Lightning	Resistance Heat-Up	NFPA 70 & 78	Paragraph 2.2.3.11
Extreme Environmental Temperature	125 ºF (50 ºC)	-	Paragraph 2.2.3.13
Load Combinations:	See Table 2.2.14	ANSI/ANS 57.9 and NUREG-1536	Subsection 2.2.7

2.1 SPENT FUEL TO BE STORED

2.1.1 Determination of the Design Basis Fuel

A central object in the design of the HI-STAR 100 System is to ensure that a majority of SNF discharged from the U.S. reactors can be loaded into one of the HI-STAR 100 MPCs. Publications such as Refs. [2.1.1] and [2.1.2] provide a comprehensive description of fuel discharged from U.S. reactors.

The cell openings and lengths in the fuel basket have been sized to accommodate BWR and PWR assemblies listed in Refs. [2.1.1] and [2.1.2] except as noted below. Similarly, the cavity length of the multi-purpose canisters has been set at a dimension which permits storing most types of PWR fuel assemblies and BWR fuel assemblies with or without fuel channels.

In addition to satisfying the cross sectional and length compatibility, the active fuel region of the SNF must be enveloped in the axial direction by the neutron absorber located in the MPC fuel basket. Alignment of the neutron absorber with the active fuel region is ensured by the use of upper and lower fuel spacers suitably designed to support the bottom and restrain the top of the fuel assembly. The spacers axially position the SNF assembly such that its active fuel region is properly aligned with the neutron absorber in the fuel basket. Figure 2.1.5 provides a pictorial representation of the fuel spacers positioning the fuel assembly active fuel region. Both the upper and lower fuel spacers are designed to perform their function under normal, off-normal, and accident conditions of storage.

Tables 2.1.3 and 2.1.4, provide the fuel characteristics of all groups of fuel assembly types determined to be acceptable for storage in the HI-STAR 100 system. Any fuel assembly that has fuel characteristics within the range of Tables 2.1.3 and 2.1.4 and meets the other limits specified in Table 2.1.6 is acceptable for storage in the HI-STAR 100 system. The *groups of fuel assembly types* presented in Tables 2.1.3 and 2.1.4 are defined *as "array/classes" as described in further detail in Chapter 6*. Table 2.1.5 lists the BWR and PWR fuel assembly designs which are found to govern for the three qualification criteria, namely reactivity, shielding, and decay heat generation. Substantiating results of analyses for the governing assembly types are presented in the respective chapters dealing with the specific qualification topic. Additional information on the design basis fuel definition is presented in the following subsections.

2.1.2 Intact SNF Specifications

Intact fuel assemblies are defined as fuel assemblies without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. The design payload for the HI-STAR 100 System is intact zircaloy clad fuel assemblies with the characteristics listed in Table 2.1.6 or intact stainless steel clad fuel assemblies with the characteristics listed in Table 2.1.11. The placement of a single stainless steel clad fuel assembly in an MPC necessitates that all fuel assemblies (stainless steel clad or zircaloy clad) stored in that MPC meet the maximum heat generation requirements for stainless steel clad fuel specified in Table 2.1.11. Intact BWR MOX fuel assemblies shall meet the requirements of Table 2.1.7.

Intact fuel assemblies with missing pins cannot be loaded into the HI-STAR 100 System unless dummy fuel pins, which occupy a volume greater than or equal to the original fuel pins, replace the missing pins prior to loading. Any intact fuel assembly which falls within the geometric, thermal, and nuclear limits established for the design basis intact fuel assembly can be safely stored in the HI-STAR 100 System.

The fuel characteristics specified in Tables 2.1.3 and 2.1.4 have been evaluated in this FSAR and are acceptable for storage in the HI-STAR 100 System.

2.1.3 Damaged SNF and Fuel Debris Specifications

Damaged fuel assemblies are defined as fuel assemblies with known or suspected cladding defects greater than pinhole leaks and hairline cracks or missing fuel rods that are not replaced with dummy fuel rods, and which may have mechanical damage which would not allow it to be handled by normal means; however, there shall be no loose components. No loose fuel debris is allowed with the damaged fuel assembly.

Fuel debris is defined as fuel assemblies with known or suspected defects greater than pinhole leaks or hairline cracks such as ruptured fuel rods, severed fuel rods, or loose fuel pellets, and which cannot be handled by normal means.

To aid in loading and unloading, damaged fuel assemblies and fuel debris will be loaded into stainless steel damaged fuel containers (DFCs) provided with mesh screens having between 40x40 and 250x250 openings per inch, for storage in the HI-STAR 100 System. This application requests approval of Dresden Unit 1 (UO_2 rods and MOX fuel rods) and Humboldt Bay fuel arrays (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A) as damaged fuel assembly contents for storage in the MPC-68 and fuel debris as contents for storage in the MPC-68F. The design characteristics bounding Dresden Unit 1 and Humboldt Bay SNF are given in Table 2.1.7. The placement of a single damaged fuel assembly in an MPC-68 or a single fuel debris damaged fuel container in an MPC-68F necessitates that all fuel assemblies (intact, damaged, or debris) stored in that MPC meet the maximum heat generation requirements specified in Table 2.1.7. The fuel characteristics specified in Table 2.1.4 for Dresden 1 and Humboldt Bay fuel arrays have been evaluated in this FSAR and are acceptable for storage as damaged fuel or fuel debris in the HI-STAR 100 System. Because of the long cooling time, small size, and low weight of spent fuel assemblies qualified as damaged fuel or fuel debris, the DFC and its contents are bounded by the structural, thermal, and shielding analyses performed for the intact BWR design basis fuel. Separate criticality analysis of the bounding fuel assembly for the damaged fuel and fuel debris has been performed in Chapter 6.

2.1.4 Structural Parameters for Design Basis SNF

The main physical parameters of a SNF assembly applicable to the structural evaluation are the fuel assembly length, envelope (cross sectional dimensions), and weight. These parameters, which define the mechanical and structural design, are listed in Tables 2.1.6, 2.1.7, and 2.1.11. The centers of gravity reported in Section 3.2 are based on the maximum fuel assembly weight. Upper and lower fuel spacers (as appropriate) maintain the axial position of the fuel assembly within the MPC basket and, therefore, the location of the center of gravity. The upper and lower fuel spacers are designed to withstand normal, off-normal, and accident conditions of storage. An axial clearance of approximately 2 to 2-1/2 inches (50 – 64 mm) is provided to account for the irradiation and thermal growth of the fuel assemblies. The suggested upper and lower fuel spacer lengths are listed in Tables 2.1.9 and 2.1.10. In order to qualify for storage in the HI-STAR 100 MPC, the SNF must satisfy the physical parameters listed in Tables 2.1.6, 2.1.7, or 2.1.11.

2.1.5 Thermal Parameters for Design Basis SNF

The principal thermal design parameter for the stored fuel is the peak fuel cladding temperature, which is a function of the maximum heat generation rate per assembly and the decay heat removal capabilities of the HI-STAR 100 System. The maximum heat generation rate per assembly for the design basis fuel assembly is based on the fuel assembly type with the highest decay heat for a given enrichment, burnup, and cooling time. This decay heat design basis fuel assembly is listed in Table 2.1.5. Section 5.2 describes the method used to determine the design basis fuel assembly type and calculate the decay heat load.

To ensure the allowable fuel cladding temperature limits are not exceeded, Table 2.1.20 specifies the allowable decay heat per assembly for Zircaloy clad fuel for the MPCs. Tables 2.1.7 and 2.1.11 provide the maximum heat generation for damaged zircaloy clad fuel assemblies and stainless steel clad fuel assemblies, respectively. Due to the conservative thermal assessment and the long cooling time of the damaged and stainless steel clad fuel, a reduction in decay heat load is not required as the cooling time increases beyond the minimum specified.

The specified decay heat load can be attained by varying burnups and cooling times. Tables 2.1.13, 2.1.14, and 2.1.15 provides illustrative burnup and cooling time characteristics for intact Zircaloy clad fuel to meet the thermal requirements for the MPC-24, MPC-32 and MPC-68. Any intact Zircaloy clad fuel assembly with a burnup and cooling time which lies below values specified in Tables 2.1.13, 2.1.14, and 2.1.15 may be thermally acceptable for loading into the HI-STAR 100 System.

The fuel rod cladding temperature is also affected by other factors. A governing geometry which maximizes the impedance to the transmission of heat out of the fuel rods has been defined. The governing thermal parameters to ensure that the range of SNF discussed previously are bounded by the thermal analysis is discussed in detail and specified in Chapter 4. By utilizing these bounding thermal parameters, the calculated peak fuel rod cladding temperatures are conservative for actual spent fuel assemblies which will have greater thermal conductivities.

Finally, the axial variation in the heat generation rate in the design basis fuel assembly is defined based on the axial burnup distribution. For this purpose, the data provided in Refs. [2.1.3] and [2.1.4] are utilized and summarized in Table 2.1.8 for reference. These distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STAR 100 System.

2.1.6 Radiological Parameters for Design Basis SNF

The principal radiological design criteria for the HI-STAR 100 System are the 10CFR72.104 site boundary dose rate limits and maintaining operational dose rates as low as reasonably achievable (ALARA). The radiation dose is directly affected by the gamma and neutron source terms of the SNF assembly.

The gamma and neutron sources are separate and are affected differently by enrichment, burnup, and cooling time. It is recognized that, at a given burnup, the radiological source terms increase monotonically as the initial enrichment is reduced. *The shielding analyses presented in Chapter 5 use the burnup and cooling time combinations which are either equal to or conservatively bound the acceptable burnup levels and cooling times shown in Tables 2.1.11, and 2.1.13 through 2.1.15*. The shielding design basis fuel assemblies thus bound all other fuel assemblies.

The design basis dose rates can be met by a variety of burnup levels and cooling times. Tables 2.1.6, 2.1.7, 2.1.11 and 2.1.14 provide the burnup and cooling time values which meet the radiological source term requirements for BWR fuel in MPC-68, respectively. *Tables 2.1.6, 2.1.11 and 2.1.13 provides the burnup and cooling time values which meet the radiological source term requirements for PWR fuel in MPC-24. Tables 2.1.6 and 2.1.15 provides the burnup and cooling time values which meet the radiological source term requirements for PWR fuel in MPC-32.*

Table 2.1.8 provides the axial distribution for the radiological source terms for PWR and BWR fuel assemblies based on the axial burnup distribution. The axial burnup distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STAR 100 System.

Thoria rods placed in Dresden Unit 1 Thoria Rod Canisters meeting the requirements of Table 2.1.12 and Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source have been qualified for storage. Up to one Dresden Unit 1 Thoria Rod Canister plus any combination of damaged fuel assemblies in damaged fuel containers and intact fuel, up to a total of 68 may be stored.

Burnable Poison Rod Assemblies (BPRAs) and Thimble Plug Devices (TPDs) in PWR fuel have been qualified for storage in the MPC-24.

2.1.7 Criticality Parameters for Design Basis SNF

As discussed earlier, the MPC-68, MPC-68F and MPC-32 all feature a basket without flux traps. In the aforementioned baskets, there is one panel of neutron absorber between two adjacent fuel assemblies. The MPC-24 employs a construction wherein two neighboring fuel assemblies are separated by two panels of neutron absorber with a water gap between them (flux trap construction).

The minimum ^{10}B areal density in the neutron absorber panels for each MPC model is shown in Table 2.1.17.

For all MPCs, the ^{10}B areal density used for criticality analysis is conservatively established below *the minimum values shown in Table 2.1.17. For Boral, the value used in the analysis is 75% of the minimum value, while for METAMIC, it is 90% of the minimum value. This is consistent with NUREG-1536 [2.1.5], which requires a 25% reduction in ^{10}B areal density credit when subject to standard acceptance tests, and which allows a smaller reduction when more comprehensive tests of the areal density are performed.*

The criticality analyses for the MPC-24 and for the MPC-32 were performed with credit taken for soluble boron in the MPC water during wet loading and unloading operations. Table 2.1.18 and 2.1.19 provide the required soluble boron concentrations for these MPCs. The MPC-24 may be used for PWR fuel without taking any soluble boron credit if the enrichment limit is met.

2.1.8 Summary of SNF Design Criteria

An intact zircaloy clad fuel assembly is acceptable for storage in a HI-STAR 100 System if it fulfills the following criteria:

a. It satisfies the physical characteristics listed in Tables 2.1.3 or 2.1.4, and 2.1.6.

b. Its initial enrichment is less than that indicated by Table 2.1.6 for the MPC it is intended to be stored in.

c. The period from discharge is greater than or equal to the minimum cooling time listed in Table 2.1.6, and the decay heat is equal to or less than the value stated in Table 2.1.20 for a given cooling time.

d. The average burnup of the fuel assembly is equal to or less than the burnup specified in Tables 2.1.13, 2.1.14, and 2.1.15 for a given cooling time.

A damaged fuel assembly shall meet the characteristics specified in Table 2.1.7 for storage in the MPC-68. Fuel debris shall meet the characteristics specified in Table 2.1.7 for storage in the MPC-68F.

Stainless steel clad fuel assemblies shall meet the characteristics specified in Table 2.1.11 for storage in the MPC-24 or MPC-68.

MOX BWR fuel assemblies shall meet the requirements of Tables 2.1.6 and 2.1.7 for intact and damaged fuel/fuel debris, respectively.

Only control components specifically the tables in Section 2.1 for PWR fuel are to be included with the fuel assembly. Burnable Poison Rod Assemblies (BPRAs) and Thimble Plug Devices (TPDs) are authorized for storage in the MPC-24. Fuel assemblies with BPRAs and TPDs shall meet the burnup and cooling time requirements specified in Tables 2.1.13, 2.1.15 and 2.1.16.

Fuel with non-zircaloy grid spacers are only permitted in MPC-32, with burnup and cooling time requirements specified in Table 2.1.15.

Thoria rods placed in Dresden Unit 1 Thoria Rod Canisters meeting the requirements of Table 2.1.12 are authorized for storage. Up to one Dresden Unit 1 Thoria Rod Canister plus any combination of damaged fuel assemblies in damaged fuel containers and intact fuel, up to a total of 68 may be stored.

Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68 or MPC-68F.

Table 2.1.1

INTENTIONALLY DELETED

Table 2.1.2

INTENTIONALLY DELETED

Table 2.1.3
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**14x14 A**	**14x14 B**	**14x14 C**	**14x14 D**
Clad Material (Note 2)	Zr	Zr	Zr	SS
Design Initial U (kg/assy.) (Note 3)	≤ 407	≤ 407	≤ 425	≤ 400
Initial Enrichment *(MPC-24 without soluble boron credit) (wt % ^{235}U)*	≤ 4.6	≤ 4.6	≤ 4.6	≤ 4.0
Initial Enrichment (MPC-24 and MPC-32 with soluble boron credit – See Note 6) (wt % ^{235}U)	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*
No. of Fuel Rods (Note 5)	179	179	176	180
Fuel Clad O.D. in (mm)	≥ 0.400 (10.16)	≥ 0.417 (10.52)	≥ 0.440 (11.18)	≥ 0.422 (10.72)
Fuel Clad I.D. in (mm)	≤ 0.3514 (8.93)	≤ 0.3734 (9.48)	≤ 0.3880 (9.86)	≤ 0.3890 (9.88)
Fuel Pellet Dia. in (mm) (Note 7)	≤ 0.3444 (8.75)	≤ 0.3659 (9.29)	≤ 0.3805 (9.66)	≤ 0.3835 (9.74)
Fuel Rod Pitch in (mm)	≤ 0.556 (14.1)	≤ 0.556 (14.1)	≤ 0.580 (14.7)	≤ 0.556 (14.1)
Active Fuel Length in (mm)	≤ 150 (3810)	≤ 150 (3810)	≤ 150 (3810)	≤ 144 (3658)
No. of Guide Tubes	17	17	5 (Note 4)	16
Guide Tube Thickness in (mm)	≥ 0.017 (0.43)	≥ 0.017 (0.43)	≥ 0.038 (0.97)	≥ 0.0145 (0.37)

Table 2.1.3 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**15x15 A**	**15x15 B**	**15x15 C**	**15x15 D**	**15x15 E**	**15x15 F**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 464	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment *(MPC-24 without soluble boron credit) (wt % ^{235}U)*	≤ 4.1	≤ 4.1	≤ 4.1	≤ 4.1	≤ 4.1	≤ 4.1
Initial Enrichment (MPC-24 and MPC-32 with soluble boron credit – Note 6) (wt % ^{235}U)	≤ 5.0	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*
No. of Fuel Rods (Note 5)	204	204	204	208	208	208
Fuel Clad O.D. in (mm)	≥ 0.418 (10.6)	≥ 0.420 (10.7)	≥ 0.417 (10.6)	≥ 0.430 (10.9)	≥ 0.428 (10.9)	≥ 0.428 (10.9)
Fuel Clad I.D. in (mm)	≤ 0.3660 (9.3)	≤ 0.3736 (9.49)	≤ 0.3640 (9.25)	≤ 0.3800 (9.65)	≤ 0.3790 (9.63)	≤ 0.3820 (9.7)
Fuel Pellet Dia. in (mm) (Note 7)	≤ 0.3580 (9.09)	≤ 0.3671 (9.32)	≤ 0.3570 (9.07)	≤ 0.3735 (9.49)	≤ 0.3707 (9.42)	≤ 0.3742 (9.5)
Fuel Rod Pitch in (mm)	≤ 0.550 (14)	≤ 0.563 (14.3)	≤ 0.563 (14.3)	≤ 0.568 (14.4)	≤ 0.568 (14.4)	≤ 0.568 (14.4)
Active Fuel Length in (mm)	≤ 150 (3810)	≤ 150 (3810)	≤ 150 (3810)	≤ 150 (3810)	≤ 150 (3810)	≤ 150 (3810)
No. of Guide Tubes	21	21	21	17	17	17
Guide Tube Thickness in (mm)	≥ 0.0165 (0.42)	≥ 0.015 (0.38)	≥ 0.0165 (0.42)	≥ 0.0150 (0.38)	≥ 0.0140 (0.36)	≥ 0.0140 (0.36)

Table 2.1.3 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**15x15 G**	**15x15 H**	**16x16 A**	**17x17A**	**17x17 B**	**17x17 C**
Clad Material (Note 2)	SS	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	$\leq$ 420	$\leq$ 475	$\leq$ 443	$\leq$ 467	$\leq$ 467	$\leq$ 474
Initial Enrichment *(MPC-24 without soluble boron credit) (wt % ^{235}U)*	$\leq$ 4.0	$\leq$ 3.8	$\leq$ 4.6	$\leq$ 4.0	$\leq$ 4.0	$\leq$ 4.0
Initial Enrichment (MPC-24 and MPC-32 with soluble boron credit – See Note 6) (wt % ^{235}U)	*$\leq$ 5.0*	*$\leq$ 5.0*	*$\leq$ 5.0*	*$\leq$ 5.0*	*$\leq$ 5.0*	*$\leq$ 5.0*
No. of Fuel Rods (Note 5)	204	208	236	264	264	264
Fuel Clad O.D. in (mm)	$\geq$ 0.422 (10.72)	$\geq$ 0.414 (10.52)	$\geq$ 0.382 (9.70)	$\geq$ 0.360 (9.14)	$\geq$ 0.372 (9.45)	$\geq$ 0.377 (9.58)
Fuel Clad I.D. in (mm)	$\leq$ 0.3890 (9.88)	$\leq$ 0.3700 (9.4)	$\leq$ 0.3320 (8.43)	$\leq$ 0.3150 (8)	$\leq$ 0.3310 (8.41)	$\leq$ 0.3330 (8.46)
Fuel Pellet Dia. in (mm) *(Note 7)*	$\leq$ 0.3825 (9.72)	$\leq$ 0.3622 (9.2)	$\leq$ 0.3255 (8.27)	$\leq$ 0.3088 (7.84)	$\leq$ 0.3232 (8.21)	$\leq$ 0.3252 (8.26)
Fuel Rod Pitch in (mm)	$\leq$ 0.563 (14.3)	$\leq$ 0.568 (14.43)	$\leq$ 0.506 (12.85)	$\leq$ 0.496 (12.6)	$\leq$ 0.496 (12.6)	$\leq$ 0.502 (12.75)
Active Fuel Length in (mm)	$\leq$ 144 (3658)	$\leq$ 150 (3810)	$\leq$ 150 (3810)	$\leq$ 150 (3810)	$\leq$ 150 (3810)	$\leq$ 150 (3810)
No. of Guide Tubes	21	17	5 (Note 4)	25	25	25
Guide Tube Thickness in (mm)	$\geq$ 0.0145 (0.37)	$\geq$ 0.0140 (0.36)	$\geq$ 0.0400 (1)	$\geq$ 0.016 (0.41)	$\geq$ 0.014 (0.36)	$\geq$ 0.020 (0.51)

Table 2.1.3 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS

NOTES:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

2. *See Table 1.0.1 for the definition of "ZR".*

3. Design initial uranium weight is the *nominal* uranium weight specified for each fuel assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer tolerances.

4. Each guide tube replaces four fuel rods.

5. Missing fuel rods must be replaced with dummy fuel rods that displace an equal or greater amount of water as the original fuel rods.

6. *Soluble boron concentration per Table 2.1.18 and 2.1.19, as applicable.*

7. *Annular fuel pellets are allowed in the top and bottom 12" (304.8 mm) of the active fuel length.*

Table 2.1.4
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**6x6 A**	**6x6 B**	**6x6 C**	**7x7 A**	**7x7 B**	**8x8 A**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 195	≤ 120
Maximum Planar-Average Initial Enrichment (wt. % ^{235}U)	≤ 2.7	≤ 2.7 for the UO_2 rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt. % ^{235}U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rods	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ .4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	≥0	≥0	N/A	N/A	N/A	≥0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table 2.1.4 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**8x8 B**	**8x8 C**	**8x8 D**	**8x8 E**	**8x8 F**	**9x9 A**	**9x9 B**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 185	≤ 185	≤ 185	≤ 185	≤ 185	≤ 177	≤ 177
Maximum Planar-Average Initial Enrichment (wt. % ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 3.6	≤ 4.2	≤ 4.2
Initial Maximum Rod Enrichment (wt. % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	63 or 64	62	60 or 61	59	64	74/66 (Note 5)	72
Fuel Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400	≥ 0.4330
Fuel Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840	≤ 0.3810
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760	≤ 0.3740
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2	1 (Note 6)
Water Rod Thickness (in.)	≥ 0.034	≥ 0.00	≥ 0.00	≥ 0.034	≥ 0.0315	≥ 0.00	≥ 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120	≤ 0.120

Table 2.1.4 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**9x9 C**	**9x9 D**	**9x9 E (Note 13)**	**9x9 F (Note 13)**	**10x10A**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 177	≤ 177	≤ 177	≤ 177	≤ 186
Maximum Planar-Average Initial Enrichment (wt. % ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.1	≤ 4.1	≤ 4.2
Initial Maximum Rod Enrichment (wt. % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	80	79	76	76	92/78 (Note 8)
Fuel Clad O.D. (in.)	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4040
Fuel Clad I.D. (in.)	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3520
Fuel Pellet Dia. (in.)	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3455
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.510
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1	2	5	5	2
Water Rod Thickness (in.)	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.030
Channel Thickness (in.)	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

Table 2.1.4 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**10x10 B**	**10x10 C**	**10x10 D**	**10x10 E**
Clad Material (Note 2)	Zr	Zr	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 186	≤ 186	≤ 125	≤ 125
Maximum Planar-Average Initial Enrichment (wt. % ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt. % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	91/83 (Note 9)	96	100	96
Fuel Clad O.D. (in.)	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Clad I.D. (in.)	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table 2.1.4 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS

NOTES:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

2. *See Table 1.0.1 for the definition of "ZR".*

3. Design initial uranium weight is the *nominal* uranium weight specified for each fuel assembly by the fuel manufacturer or reactor user. For each *BWR* fuel assembly, the total uranium weight limit specified in this table may be increased up to 1.5 percent for comparison with users' fuel records to account for manufacturer tolerances.

4. $\leq$ 0.635 wt. % ^{235}U and $\leq$ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu), (wt. % of total fuel weight, i.e., UO_2 plus PuO_2).

5. This assembly class contains 74 total fuel rods; 66 full length rods and 8 partial length rods.

6. Square, replacing nine fuel rods.

7. Variable.

8. This assembly class contains 92 total fuel rods; 78 full length rods and 14 partial length rods.

9. This assembly class contains 91 total fuel rods; 83 full length rods and 8 partial length rods.

10. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water *rods* dividing the assembly into four quadrants.

11. These rods may *also* be sealed at both ends and contain Zr material in lieu of water.

12. This assembly is known as "QUAD+". *It* has four rectangular water cross segments dividing the assembly into four quadrants.

13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or the 9x9F set of limits or clad O.D., clad I.D., and pellet diameter.

Table 2.1.5

DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION

Criterion	**MPC-68**	**MPC-24 & MPC-32**
Reactivity (Criticality)	GE12/14 10x10 with Partial Length Rods (Class 10x10A)	B&W 15x15 (Class 15x15F)
Source Term (Shielding)	GE 7x7 (Class 7x7B)	B&W 15x15 (Class 15x15F)
Decay Heat (Thermal-Hydraulic)	*GE-11 9x9*	*W 17x17 OFA*

Table 2.1.6
CHARACTERISTICS FOR DESIGN BASIS INTACT ZIRCALOY CLAD FUEL ASSEMBLIES

	MPC-68	**MPC-24 & MPC-32**
PHYSICAL PARAMETERS:		
Max. assembly width† in. (mm)	5.85	8.54 (217)
Max. assembly length† in. (mm)	176.2	176.8 (4490.7)
Max. assembly weight lb. (kg)	700	1680 (762)
Max. active fuel length† in. (mm)	150	150 (3810)
Fuel rod clad material	Zircaloy	Zircaloy
RADIOLOGICAL AND THERMAL CHARACTERISTICS:		
	MPC-68	**MPC-24 & MPC-32**
Max. initial enrichment (wt% ^{235}U)	4.2 2.7 (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A)	See Table 2.1.3
Max. heat generation (W)	Table 2.1.20 115 (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A) 183.5 (Assembly Class 8x8F)	Table 2.1.20
Max. average burnup (MWD/MTU)	See Table 2.1.14 30,000 (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A) 27,500 (Assembly Class 8x8F)	See Tables 2.1.13 and 2.1.15
Min. cooling time (years)	See Table 2.1.14 18 (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A) 10 (Assembly Class 8x8F)	See Tables 2.1.13 and 2.1.15

† Unirradiated nominal design dimensions are shown.

Table 2.1.7

DESIGN CHARACTERISTICS FOR DAMAGED ZIRCALOY CLAD FUEL ASSEMBLIES AND BWR ZIRCALOY CLAD FUEL DEBRIS

	MPC-68 (Damaged Fuel)	MPC-68F (Fuel Debris)
PHYSICAL PARAMETERS:		
Max. assembly width† (in.)	4.7	4.7
Max. assembly length† (in.)	135	135
Max. assembly weight†† (lb.)	400	400
Max. active fuel length† (in.)	110	110
Fuel rod clad material	Zircaloy	Zircaloy
RADIOLOGICAL AND THERMAL CHARACTERISTICS:		
Max. heat generation (W)	115	115
Min. cooling time (yr)	18	18
Max. initial enrichment (w/o ^{235}U) for UO_2 rods	2.7	2.7
Max. initial enrichment for MOX rods	0.612 wt.% ^{235}U 1.578 wt. % Total Fissile Plutonium	0.612 wt.% ^{235}U 1.578 wt. % Total Fissile Plutonium
Max. average burnup (MWD/MTU)	30,000	30,000

Note:

1. A maximum of four (4) damaged fuel containers with BWR zircaloy clad fuel debris may be stored in the MPC-68F with the remaining locations filled with undamaged or damaged fuel assemblies meeting the maximum heat generation specifications of this table.

† Dimensions envelop unirradiated nominal dimensions of Array/Class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A (Dresden Unit 1 and Humboldt Bay SNF).

†† Fuel assembly weight including hardware based on DOE MPC DPS [2.1.6]. Weight does not include damaged fuel container.

Table 2.1.8

NORMALIZED DISTRIBUTION BASED ON BURNUP PROFILE

PWR DISTRIBUTION[†]		
Interval	**Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)**	**Normalized Distribution**
1	0% to 4-1/6%	0.5485
2	4-1/6% to 8-1/3%	0.8477
3	8-1/3% to 16-2/3%	1.0770
4	16-2/3% to 33-1/3%	1.1050
5	33-1/3% to 50%	1.0980
6	50% to 66-2/3%	1.0790
7	66-2/3% to 83-1/3%	1.0501
8	83-1/3% to 91-2/3%	0.9604
9	91-2/3% to 95-5/6%	0.7338
10	95-5/6% to 100%	0.4670
BWR DISTRIBUTION[††]		
Interval	**Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)**	**Normalized Distribution**
1	0% to 4-1/6%	0.2200
2	4-1/6% to 8-1/3%	0.7600
3	8-1/3% to 16-2/3%	1.0350
4	16-2/3% to 33-1/3%	1.1675
5	33-1/3% to 50%	1.1950
6	50% to 66-2/3%	1.1625
7	66-2/3% to 83-1/3%	1.0725
8	83-1/3% to 91-2/3%	0.8650
9	91-2/3% to 95-5/6%	0.6200
10	95-5/6% to 100%	0.2200

† Reference 2.1.3

†† Reference 2.1.4

Table 2.1.9

SUGGESTED PWR UPPER AND LOWER FUEL SPACER LENGTHS

Fuel Assembly Type	**Assembly Length w/o C.C.† in. (mm)**	**Location of Active Fuel from Bottom in. (mm)**	**Max. Active Fuel Length in. (mm)**	**Upper Fuel Spacer Length in. (mm)**	**Lower Fuel Spacer Length in. (mm)**
CE 14x14	157 (3990)	4.1 (100)	137 (3480)	9.5 (240)	10.0 (255)
CE 16x16	176.8 (4490)	4.7 (120)	150 (3810)	0 (0)	0 (0)
BW 15x15	165.7 (4210)	8.4 (210)	141.8 (3600)	6.7 (170)	4.1 (100)
W 17x17 OFA	159.8 (4060)	3.7 (95)	144 (3660)	8.2 (210)	8.5 (215)
W 17x17 Std	159.8 (4060)	3.7 (95)	144 (3660)	8.2 (210)	8.5(215)
W 17x17 V5H	160.1 (4070)	3.7 (95)	144 (3660)	7.9 (200)	8.5(215)
W 15x15	159.8 (4060)	3.7 (95)	144 (3660)	8.2 (210)	8.5(215)
W 14x14 Std	159.8 (4060)	3.7 (95)	145.2 (3690)	9.2 (230)	7.5(190)
W 14x14 OFA	159.8 (4060)	3.7 (95)	144 (3660)	8.2 (210)	8.5(215)
Ft. Calhoun	146 (3710)	6.6 (170)	128 (3250)	10.25 (260)	20.25(510)
St. Lucie 2	158.2 (4020)	5.2 (130)	136.7 (3470)	10.25 (260)	8.05(205)
B&W 15x15 SS	137.1 (3480)	3.873 (100)	120.5 (3060)	19.25 (490)	19.25(490)
W 15x15 SS	137.1(3480)	3.7 (95)	122 (3100)	19.25 (490)	19.25(490)
W 14x14 SS	137.1(3480)	3.7 (95)	120 (3050)	19.25 (490)	19.25(490)

Note: *Each user shall specify the fuel spacer length based on their fuel assembly length, presence of a DFC, and allowing an approximate 2 to 2-1/2-inch (50- 64 mm) gap under the MPC lid. Fuel spacers shall be sized to ensure that the active fuel region of intact fuel assemblies remains within the neutron poison region of the MPC basket with water in the MPC.*

† C.C. is an abbreviation for Control Components. Fuel assemblies with control components may require shorter fuel spacers. Each user shall specify the fuel spacer lengths based on their fuel length and any control components and allowing an approximate 2-inch (50 mm) gap.

Table 2.1.10

SUGGESTED BWR UPPER AND LOWER FUEL SPACER LENGTHS

Fuel Assembly Type	**Assembly Length (in.)**	**Location of Active Fuel from Bottom (in.)**	**Max. Active Fuel Length (in.)**	**Upper Fuel Spacer Length (in.)**	**Lower Fuel Spacer Length (in.)**
GE/2-3	171.2	7.3	150	4.8	0
GE/4-6	176.2	7.3	150	0	0
Dresden 1	134.4	11.2	110	18.0	28.0
Humboldt Bay	95.0	8.0	79.0	40.5	40.5
Dresden 1 Damaged Fuel or Fuel Debris	142.1†	11.2	110.0	17.0	16.9
Humboldt Bay Damaged Fuel or Fuel Debris	105.5†	8.0	79.0	35.25	35.25
LaCrosse	102.5	10.5	83.0	37.0	37.5

Note: *Each user shall specify the fuel spacer length based on their fuel assembly length, presence of a DFC, and allowing an approximate 2 to 2-1/2-inch gap under the MPC lid. Fuel spacers shall be sized to ensure that the active fuel region of intact fuel assemblies remains within the neutron poison region of the MPC basket with water in the MPC.*

† Fuel assembly length includes the damaged fuel container.

Table 2.1.11

DESIGN CHARACTERISTICS FOR STAINLESS STEEL CLAD FUEL ASSEMBLIES

	BWR MPC-68	PWR MPC-24
Max. assembly width† (in.)	5.62	8.42
Max. assembly length† (in.)	102.5	138.8
Max. assembly weight†† (lb.)	400	1421
Max. active fuel length† (in.)	83	122
Max. heat generation (W)	95	575 (MPC-24)
Min. cooling time (yr)	10	9 at 30,000 MWD/MTU (MPC-24) 15 at 40,000 MWD/MTU (MPC-24)
Max. initial enrichment (wt.% ^{235}U)	4.0	4.0
Max. average burnup (MWD/MTU)	22,500	40,000

† Dimensions are unirradiated nominal dimensions.
†† Fuel assembly weight including hardware based on DOE MPC DPS [2.1.6].

Table 2.1.12

DESIGN CHARACTERISTICS FOR THORIA RODS IN D1 THORIA ROD CANISTERS

PARAMETER	**MPC-68 or MPC-68F**
Cladding Type	Zircaloy (Zr)
Composition	98.5 wt.% ThO_2, 1.5 wt.% UO_2 with an enrichment of 93.5 wt. % ^{235}U
Number of Rods Per Thoria Canister	$\leq$ 18
Decay Heat Per Thoria Canister	$\leq$ 115 watts
Post-Irradiation Fuel Cooling Time and Average Burnup Per Thoria Canister	Cooling time $\geq$ 18 years and average burnup $\geq$ 16,000 MWD/MTIHM
Initial Heavy Metal Weight	$\leq$ 27 kg/canister
Fuel Cladding O.D.	$\geq$ 0.412 inches
Fuel Cladding I.D.	$\leq$ 0.362 inches
Fuel Pellet O.D.	$\leq$ 0.358 inches
Active Fuel Length	$\leq$ 111 inches
Canister Weight	$\leq$ 550 lbs., including Thoria Rods
Canister Material	*Type 304 SS*

Table 2.1.13

FUEL ASSEMBLY COOLING AND AVERAGE BURNUP (NOTE 1)

POST-IRRADIATION COOLING TIME (years)	***MPC-24 PWR Assembly Burnup (Without BPRAs and with or without TPDs)(MWD/MTU)***	***MPC-24 PWR Assembly Burnup (With BPRAs) (MWD/MTU)***
≥ 5	*≤ 28,700*	*≤ 28,300*
≥ 6	*≤ 32,700*	*≤ 32,300*
≥ 7	*≤ 33,300*	*≤ 32,700*
≥ 8	*≤ 35,500*	*≤ 35,000*
≥ 9	*≤ 37,000*	*≤ 36,500*
≥ 10	*≤ 38,200*	*≤ 37,600*
≥ 11	*≤ 39,300*	*≤ 38,700*
≥ 12	*≤ 40,100*	*≤ 39,500*
≥ 13	*≤ 40,800*	*≤ 40,200*
≥ 14	*≤ 41,500*	*≤ 40,800*
≥ 15	*≤ 42,100*	*≤ 41,400*

Note: 1. Linear interpolation between points permitted.

Table 2.1.14
FUEL ASSEMBLY COOLING AND AVERAGE BURNUP (NOTE 1)

POST-IRRADIATION COOLING TIME (years)	***MPC-68 BWR Assembly Burnup (MWD/MTU)***
≥ 5	*≤ 26,000*
≥ 6	*≤ 29,100*
≥ 7	*≤ 29,600*
≥ 8	*≤ 31,400*
≥ 9	*≤ 32,800*
≥ 10	*≤ 33,800*
≥ 11	*≤ 34,800*
≥ 12	*≤ 35,500*
≥ 13	*≤ 36,200*
≥ 14	*≤ 36,900*
≥ 15	*≤ 37,600*

Note: 1. Linear interpolation between points permitted.

Table 2.1.15

FUEL ASSEMBLY COOLING AND AVERAGE BURNUP

POST-IRRADIATION COOLING TIME (years)	***MPC-32 PWR Assembly Burnup (MWD/MTU)***	***MPC-32 PWR Assembly Burnup (Non-zircaloy Grid Spacers) (MWD/MTU)***
≥ 8	*≤ 24,500*	-
≥ 9	*≤ 29,500*	-
≥ 10	≤ 31,100	-
≥ 11	≤ 32,800	-
≥ 12	*≤ 34,500*	*≤ 24,500*
≥ 13	≤ 37,000	≤ 27,000
≥ 14	*≤ 39,500*	*≤ 29,500*
≥ 15	≤ 40,300	≤ 32,000
≥ 16	≤ 41,100	*≤ 34,500*
≥ 17	≤ 42,000	≤ 36,100
≥ 18	≤ 42,800	*≤ 39,500*
≥ 19	≤ 43,600	≤ 39,500
≥ 20	*≤ 44,500*	*≤ 42,500*

Note: 1. Linear interpolation between points permitted.

Table 2.1.16

NON-FUEL HARDWARE COOLING AND AVERAGE BURNUP (NOTE 1)

POST-IRRADIATION COOLING TIME (years)	***MPC-24 BPRA BURNUP (MWD/MTU)***	***MPC-24 TPD Burnup (MWD/MTU)***
≥ 3	*≤ 20,000*	*NC (Note 2)*
≥ 4	*NC*	*≤ 20,000*
≥ 5	*≤ 30,000*	*NC*
≥ 6	*≤ 40,000*	*≤ 30,000*
≥ 7	*NC*	*≤ 40,000*
≥ 8	*≤ 50,000*	*NC*
≥ 9	*≤ 60,000*	*≤ 50,000*
≥ 10	*NC*	*≤ 60,000*
≥ 11	*NC*	*NC*
≥ 12	*NC*	*≤ 90,000*
≥ 13	*NC*	*≤ 180,000*
≥ 14	*NC*	*≤ 630,000*

Notes: 1. Linear interpolation between points is permitted, except that TPD burnups > 180, 000 MWD/MTU and ≤ 630,000 MTD/MTU must be cooled ≥ 14 years.

2. Not Calculated

Table 2.1.17

MINIMUM BORAL ^{10}B LOADING IN NEUTRON ABSORBER PANELS

MPC MODEL	*MINIMUM ^{10}B LOADING* **(g/cm^2)**	
	Boral Neutron Absorber Panels	**METAMIC Neutron Absorber Panels**
MPC-24	*0.0267*	*0.0223*
MPC-32	*0.0372*	*0.0310*
MPC-68	*0.0372*	*0.0310*
MPC-68F	*0.01*	*N/A (Note 1)*

Notes:

1.All MPC-68F canisters are equipped with Boral neutron absorber panels.

Table 2.1.18

Soluble Boron Requirements for MPC-24 Fuel Wet Loading and Unloading Operations

MPC MODEL	***FUEL ASSEMBLY MAXIMUM AVERAGE ENRICHMENT (wt % ^{235}U)***	***MINIMUM SOLUBLE BORON CONCENTRATION (ppmb)***
MPC-24	*All fuel assemblies with initial enrichment[1] less than the prescribed value for soluble boron credit*	*0*
MPC-24	*One or more fuel assemblies with an initial enrichment[1] greater than or equal to the prescribed value for no soluble boron credit and ≤ 5.0 wt. %*	*≥ 400*

[1]Refer to Table 2.1.3 for these enrichments.

Table 2.1.19

Soluble Boron Requirements for MPC-32 Wet Loading and Unloading Operations

Fuel Assembly Array/Class	***All Intact Fuel Assemblies***	
	Max. Initial Enrichment $\leq$ 4.1 wt.% ^{235}U (ppmb)	***Max. Initial Enrichment 5.0 wt.% ^{235}U (ppmb)***
14x14A/B/C/D	*1,300*	*1,900*
15x15A/B/C/G	*1,800*	*2,500*
15x15D/E/F/H	*1,900*	*2,600*
16x16A	*1,400*	*2,000*
17x17A/B/C	*1,900*	*2,600*

Note:

1. *For maximum initial enrichments between 4.1 wt% and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be determined by linear interpolation between the minimum soluble boron concentrations at 4.1 wt% and 5.0 wt% ^{235}U.*

Table 2.1.20
ALLOWABLE FUEL ASSEMBLY COOLING AND DECAY HEAT

POST-IRRADIATION COOLING TIME (years)	***MPC-24 PWR Assembly with or without BRPAs or TPDs Decay Heat (Watts)***	***MPC-68 BWR Assembly Decay Heat (Watts)***	***MPC-32 PWR Assembly Decay Heat (Watts)***
≥ 5	*≤ 792 (Vertical)* *≤ 833 (Horizontal)*	*≤ 272*	*≤ 578 (Vertical)* *≤ 625 (Horizontal)*

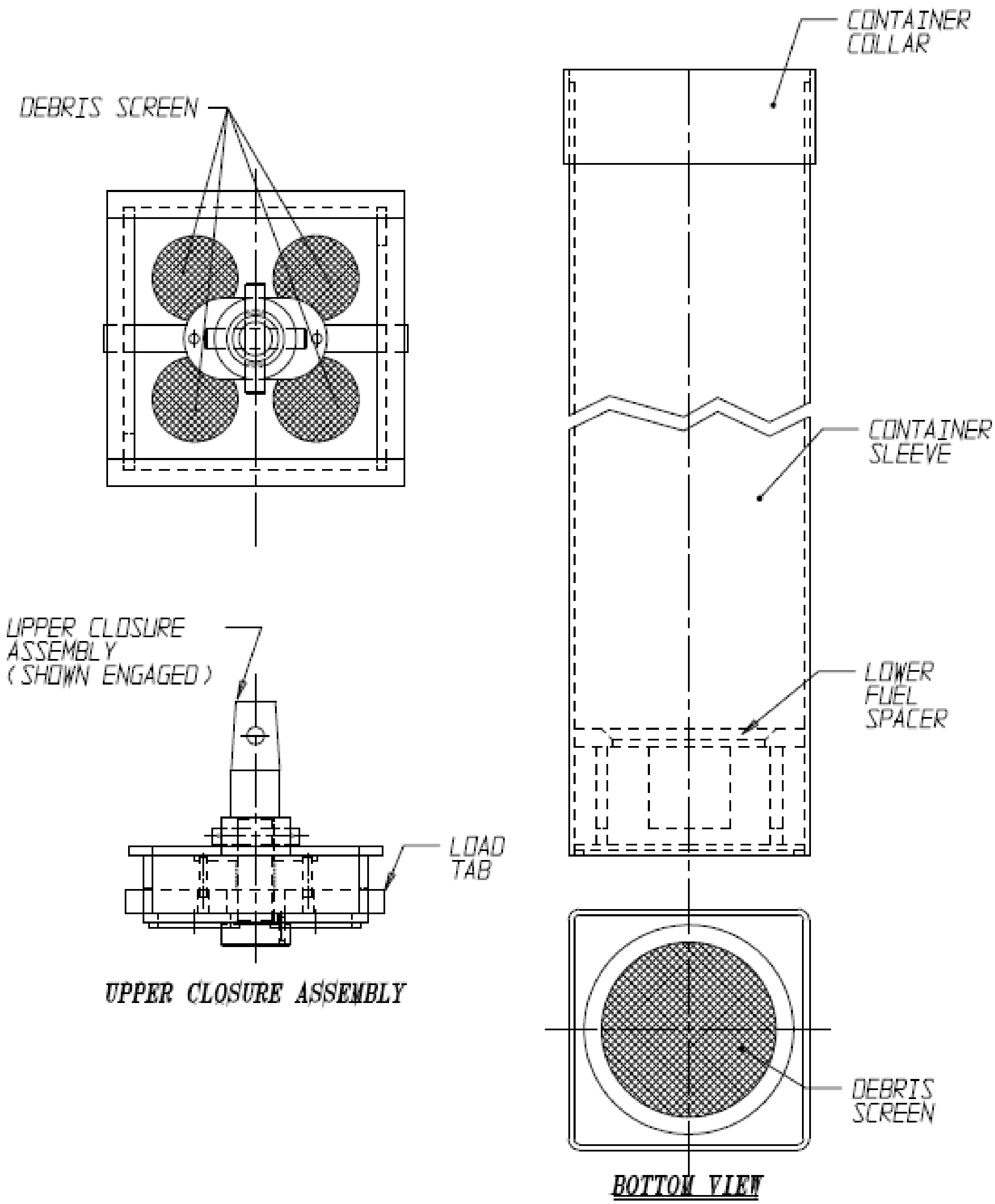

FIGURE 2.1.1; DAMAGED FUEL CONTAINER FOR DRESDEN UNIT-1/ HUMBOLDT BAY SNF

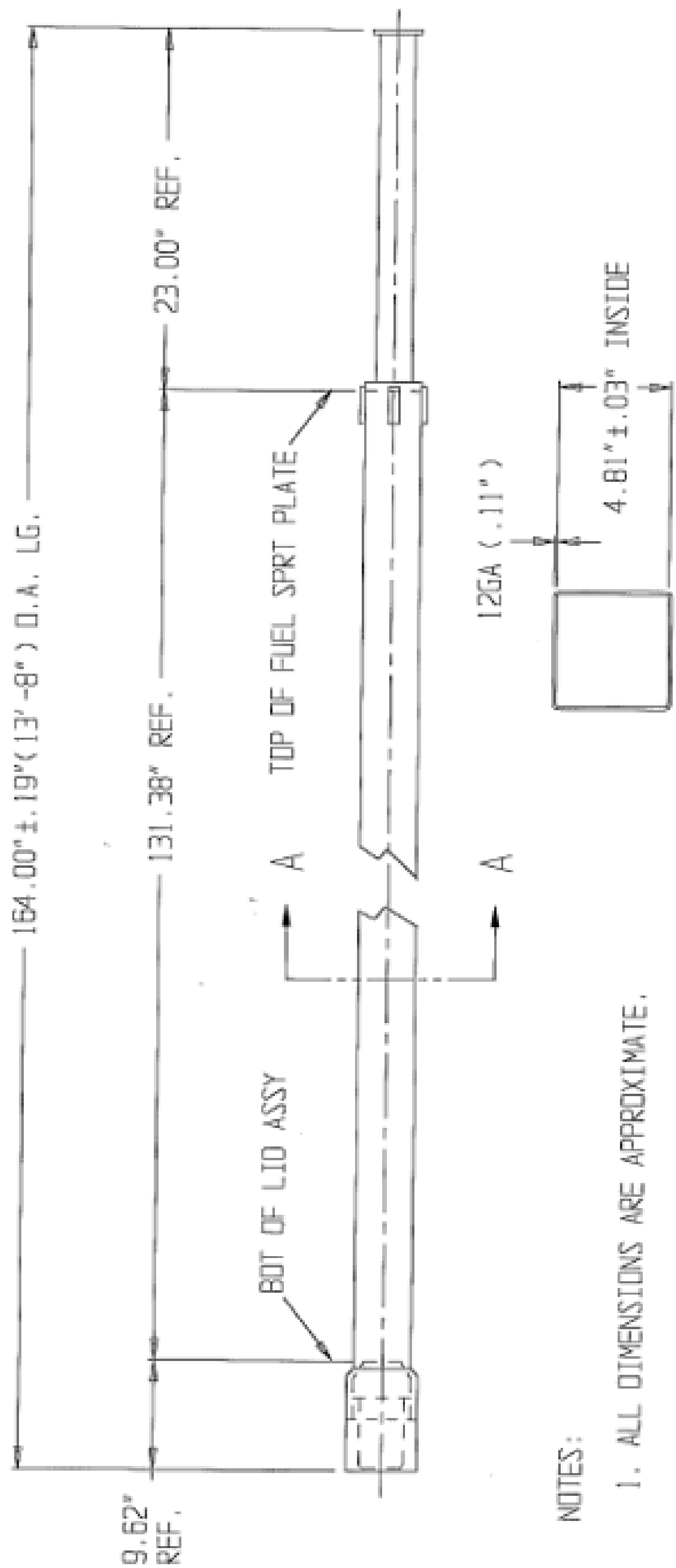

FIGURE 2.1.2; TN DAMAGED FUEL CANISTER FOR DRESDEN UNIT-1

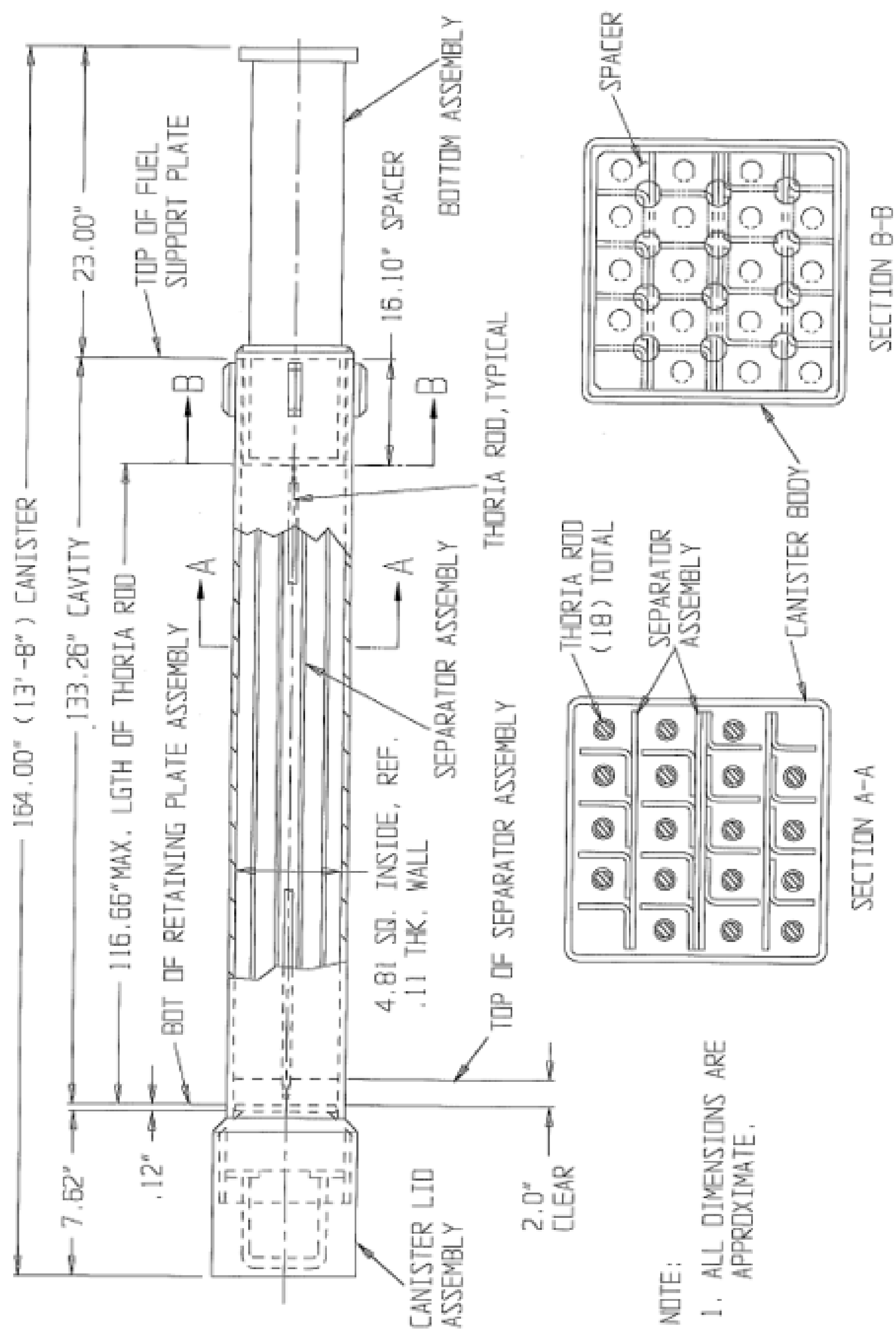

FIGURE 2.1.2A; TN THORIA ROD CANISTER FOR DRESDEN UNIT-1

FIGURE 2.1.3

INTENTIONALLY DELETED

FIGURE 2.1.4

INTENTIONALLY DELETED

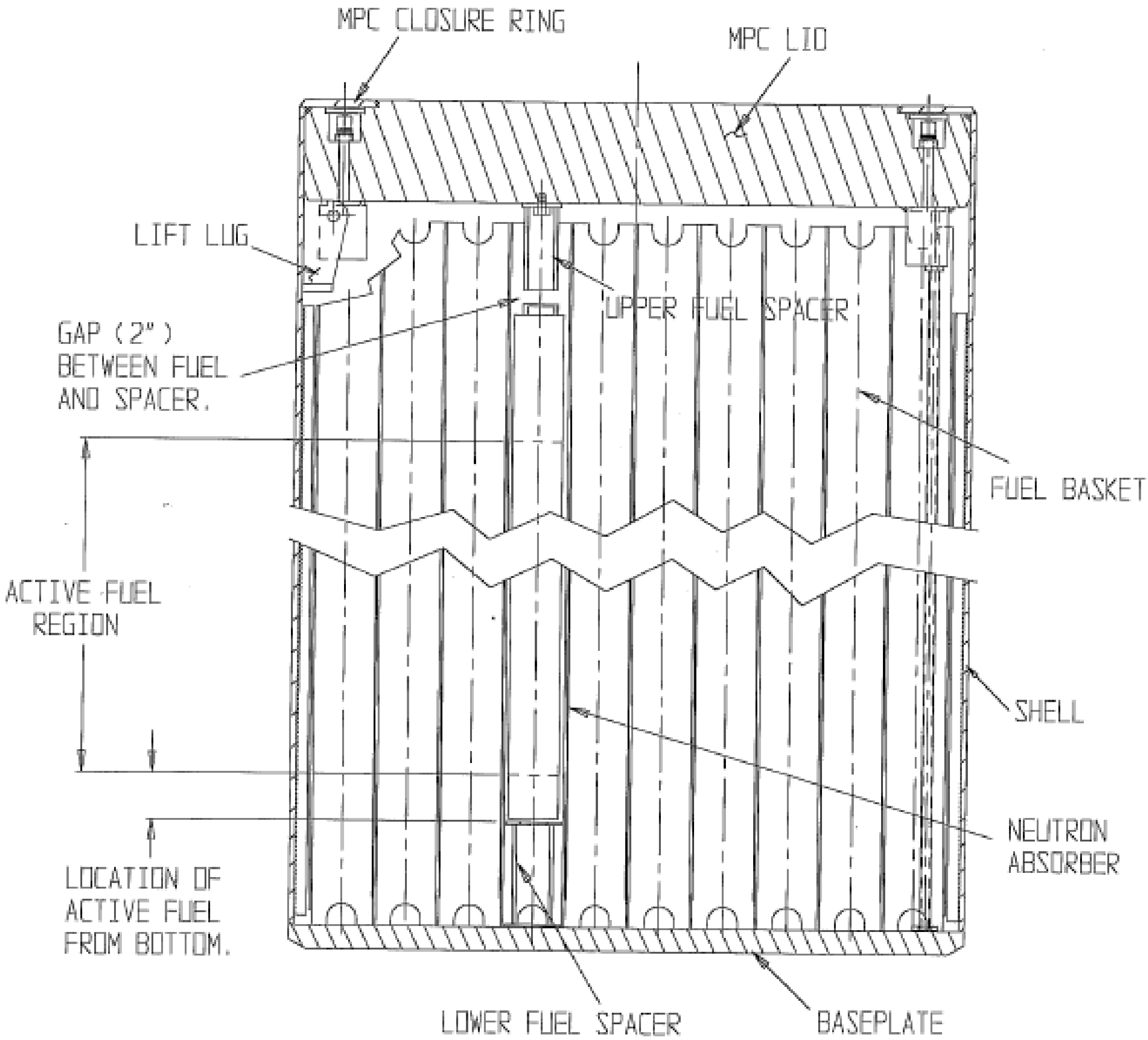

FIGURE 2.1.5; HI-STAR 100 MPC WITH UPPER AND LOWER FUEL SPACERS

FIGURE 2.1.6 through 2.1.8

INTENTIONALY DELETED

2.2 HI-STAR 100 DESIGN CRITERIA

The HI-STAR 100 System is engineered for unprotected outside storage for the duration of its design life. Accordingly, the cask system is designed to withstand normal, off-normal, and environmental phenomena or accident conditions of storage. Normal conditions include the conditions that are expected to occur regularly or frequently in the course of normal operation. Off-normal conditions include those infrequent events that could reasonably be expected to occur during the lifetime of the cask system. Environmental phenomena or accident conditions includes events that are postulated because their consideration establishes a conservative design basis.

Normal condition loads act in combination with all other loads. Off-normal condition loads and environmental phenomena or accident condition loads are not applied in combination. However, loads which occur as a result of the same phenomena are applied simultaneously. For example, the tornado wind loads are applied in combination with the tornado missile loads.

In the following subsections, the design criteria are established for normal, off-normal, and accident conditions for storage. Loads that require consideration under each condition are identified and the design criteria discussed. Based on consideration of the applicable requirements of the system, the following loads are identified:

Normal (Long-Term Storage) Condition: Dead Weight, Handling, Pressure, Temperature, Snow

Off-Normal Condition: Pressure, Temperature, Leakage of One Seal, Malfunction of Forced Helium Dehydrator System

Accident Condition: Handling Accident, Tip-Over, Fire, Partial Blockage of MPC Basket Vent Holes, Tornado, Flood, Earthquake, Fuel Rod Rupture, Confinement Boundary Leakage, Explosion, Lightning, Burial Under Debris, Extreme Environmental Temperature.

Short-Term Operations: This loading condition is defined to accord with ISG-11, Revision 3 guidance [2.0.5]. This includes those normal operational evolutions necessary to support fuel loading or unloading activities. These include, but are not limited to MPC cavity drying, helium backfill, MPC transfer.

Each of these conditions and the applicable loads are identified with applicable design criteria established. Design criteria are deemed to be satisfied if the specified allowable limits are not exceeded.

2.2.1 Normal Condition Design Criteria

2.2.1.1 Dead Weight

The HI-STAR 100 System must withstand the static loads due to the weights of each of its components.

2.2.1.2 Handling

The HI-STAR 100 System must withstand loads experienced during routine handling. *Normal handling includes:*

i. Vertical lifting and transfer to the ISFSI of the HI-STAR overpack with loaded MPC.
ii. Lifting, upending/downending, and transfer to the ISFSI of the HI-STAR with loaded MPC in the vertical or horizontal position.
iii. Lifting of the loaded MPC into and out of the HI-STAR overpack.

The loads shall be increased by 15% to include any dynamic effects from the lifting operations as directed by CMAA #70 [2.2.1].

Handling operations of the loaded HI-STAR overpack is limited to working area ambient temperatures greater than or equal to 0ºF. This limitation is specified to ensure that a sufficient safety margin exists before brittle fracture might occur during handling operations.

Lifting attachments and special lifting devices shall meet the requirements of ANSI N14.6, as applicable [1] [2.0.4].

2.2.1.3 Pressure

Pressures on the HI-STAR 100 System components depend on the bulk temperature of the helium gas and any environmental or internal factor capable of causing a pressure change. The HI-STAR 100 System must be capable of withstanding normal condition pressures.

The MPC internal pressure is dependent on the initial volume of cover gas (helium), the volume of fill gas in the fuel rods, the fraction of fission gas released from the fuel matrix, the number of fuel rods assumed to have ruptured, and temperature.

The normal condition MPC internal design pressure bounds the cumulative effects of the maximum fill gas volume, normal environmental ambient temperatures, the maximum MPC heat load, and fuel rod rupture releases per Table 2.0.1. The off-normal condition MPC internal design pressure bounds the cumulative effects of the maximum fill gas volume, off-normal environmental ambient temperatures, the maximum MPC heat load, and fuel rod rupture releases per Table 2.0.1. For conservatism, the MPC normal internal design pressure bounds both normal and off-normal conditions. Therefore, the normal and off-normal condition MPC internal pressures are set equal for analysis purposes. Table 2.2.1 provides the design pressures for the HI-STAR 100 System.

For the storage of damaged Dresden Unit 1 or Humboldt Bay BWR fuel assemblies or fuel debris (Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A) in a damaged fuel container, it is

1 Yield and ultimate strength values used in the stress compliance demonstration per ANSI N14.6 shall utilize confirmed material test data through either independent coupon testing or material suppliers= CMTR or COC, as appropriate. To ensure consistency between the design and fabrication of a lifting component, compliance with ANSI N14.6 in this FSAR implies that the guidelines of ASME Section III, Subsection NF for Class 3 structures are followed for material procurement and testing, fabrication, and for NDE during manufacturing conservatively assumed fuel rod rupture releases shall be per Table 2.0.1 for both normal and off-normal conditions. This condition is bounded by the pressure calculation for design basis intact fuel with 100% of the fuel rods ruptured in all of the fuel assemblies. It is shown in Chapter 4 that the accident condition design pressure is not exceeded with 100% of the fuel rods ruptured in all of the design basis fuel assemblies. Therefore, rupture of 100% of the fuel rods in the damaged fuel assemblies or fuel debris will not cause the MPC internal pressure to exceed the normal design pressure.

The MPC internal design pressure under accident conditions is discussed in Subsection 2.2.3.

The MPC external pressure is equivalent to the overpack internal pressure, since this pressure exists in the annulus between the MPC and the overpack. During loading of the HI-STAR 100 System, the annulus is evacuated, dried, and pressurized with helium. The helium gas in the annulus is compressed due to the difference in the thermal expansion of the MPC and the overpack when the HI-STAR 100 System has a positive heat load (See 3.4.4.2.1). Therefore, the normal and off-normal pressure specified in Table 2.2.1 is above the initial fill pressure.

The HI-STAR 100 overpack external pressure is a function of environmental conditions, which may produce a pressure loading. The normal and off-normal condition external design pressure is set at ambient standard pressure specified in Table 2.0.2.

The overpack neutron shield enclosure contains the neutron shield material Holtite-A. The enclosure is equipped with two rupture disks with a relief pressure specified in Table 9.1.5. The design temperature of the neutron shield material is set sufficiently low to ensure that under normal and off-normal condition any potential off-gassing will be negligible. However, the overpack neutron shield enclosure is designed to withstand design pressure under normal conditions. Under accident conditions, where the neutron shield material bulk temperature may exceed its design temperature, the redundant rupture disks will relieve ensuring that the pressure will not exceed the overpack neutron shield enclosure design pressure.

2.2.1.4 Environmental Temperatures

To evaluate the long-term effects of ambient temperatures on the HI-STAR 100 System, an upper bound value on the annual average ambient temperatures for the continental United States is used. The normal temperature specified in Table 2.2.2 is bounding for all reactor sites in the contiguous United States. The "normal" temperature set forth in Table 2.2.2 is intended to ensure that it is greater than the annual average of ambient temperatures at any location in the continental United States. In the northern region of the U.S., the design basis "normal" temperature used in this FSAR will be exceeded only for brief periods, whereas in the southern U.S., it may be straddled daily in summer months. Inasmuch as the sole effect of the "normal" temperature is on the computed fuel cladding temperature to establish long-term fuel integrity, it should not lie below the time averaged yearly mean for the ISFSI site. Previously licensed cask systems have employed lower "normal" temperatures (viz. 75 ^{o}F (24 ^{o}C) in Docket 72-1007) by utilizing national meteorological data.

Confirmation of the site-specific annual average ambient temperature is to be performed by the licensee, in accordance with 10CFR72.212. The annual average temperature is combined with insolation in accordance with 10CFR71.71 averaged over 24 hours in to establish the normal condition temperatures in the HI-STAR 100 System.

2.2.1.5 Design Temperatures

The ASME Boiler and Pressure Vessel Code (ASME Code) requires that the value of the vessel design temperature be established with appropriate consideration for the effect of heat generation internal or external to the vessel. The decay heat load from the spent nuclear fuel is the internal heat generation source for the HI-STAR 100 System. The ASME Code (Section III, Paragraph NCA-2142) requires the Design Temperature to be set at or above the maximum through thickness mean metal temperature of the pressure part under normal service (Level A) condition. Consistent with the terminology of NUREG-1536, this temperature is referred to as the "Design Temperature for Normal Conditions". Conservative calculations of the steady-state temperature field in the HI-STAR 100 System, under assumed environmental normal temperatures with the maximum decay heat load, result in HI-STAR component temperatures below the normal condition design temperatures for the HI-STAR 100 System defined in Table 2.2.3.

Maintaining fuel rod cladding integrity is also a design consideration. The maximum fuel rod cladding temperature limits for normal conditions are based on ISG-11 Rev 3 [2.0.5].

2.2.1.6 Snow and Ice

The HI-STAR 100 System must be capable of withstanding pressure loads due to snow and ice. ASCE 7-88 (formerly ANSI A58.1) [2.2.3] provides empirical formulas and tables to compute the effective design pressure on the overpack due to the accumulation of snow for the contiguous U.S. and Alaska. Typical calculated values for heated structures such as the HI-STAR 100 System range from 50 to 70 pounds per square foot (245 to 300 kg/m^2). For conservatism, the snow pressure loading is set at a level in Table 2.2.8, which bounds the ASCE 7-88 recommendation.

2.2.2 Off-Normal Conditions Design Criteria

As the HI-STAR 100 System is passive, loss of power and instrumentation failures are not defined as off-normal conditions. Off-normal condition design criteria are defined in the following subsections.

A discussion of the effects of each off-normal condition is provided in Subsection 11.1. Section 11.1 also provides the corrective action for each off-normal condition. The location of the detailed analysis for each event is referenced in Section 11.1.

2.2.2.1 Pressure

The HI-STAR 100 System must withstand loads due to off-normal pressure. *The off-normal condition MPC internal design pressure bounds the cumulative effects of the maximum fill gas volume, off-normal environmental ambient temperatures, the maximum MPC heat load, and* fuel rod rupture releases per Table 2.0.1.

2.2.2.2 Environmental Temperatures

The HI-STAR 100 System must withstand off-normal environmental temperatures. The off-normal environmental temperatures are specified in Table 2.2.2. The lower bound temperature occurs with no solar loads and the upper bound temperature occurs with steady-state insolation. Each bounding temperature is assumed in the analysis to persist for a duration sufficient to allow the system to reach steady-state temperatures.

Limits on the peaks in the time-varying ambient temperature at an ISFSI site are recognized in the FSAR in the specification of the off-normal temperatures. The lower bound off-normal temperature is defined as the minimum of the 72-hour average of the ambient temperature at an ISFSI site. Likewise, the upper bound off-normal temperature is defined by the maximum of 72-hour average of the ambient temperature. The lower and upper bound off-normal temperatures listed in Table 2.2.2 are intended to cover all ISFSI sites in the continental U.S. The 72-hour average temperature used in the definition of the off-normal temperature recognizes the considerable thermal inertia of the HI-STAR 100 System which reduces the effect of undulations in instantaneous temperature on the internals of the MPC.

2.2.2.3 Design Temperatures

In addition to the normal condition design temperature, we also define an "off-normal/accident condition temperature" pursuant to the provisions of NUREG-1536 and Regulatory Guide 3.61. This is, in effect, the short-term temperature which may exist during a transition state or a transient event (examples of such instances are short-term temperature excursion during canister vacuum drying and backfilling operations (transition state) and fire (transient event)). The off-normal/accident design temperatures of Table 2.2.3 are set down to bound the maximax (maximum in time and space) value of the thru-thickness average temperature of the structural or non-structural part, as applicable, during a short-term event. These enveloping values, therefore, will bound the maximum temperature reached anywhere in the part, excluding skin effects during or immediately after, a short-term event.

The off-normal/accident design temperatures for stainless steel and carbon steel components are chosen such that the material's ultimate tensile strength does not fall below 30% of its room temperature value, based on data in published references [2.2.14 and 2.2.15]. This ensures that the material will not fail due to creep rupture during these short duration transient events.

2.2.2.4 Leakage of One Seal

The MPC enclosure vessel does not contain gaskets or seals: All confinement boundary closure locations are welded. Because the material of construction (Alloy X, see Appendix 1.A) is known from extensive industrial experience to lend to high integrity, high ductility and high fracture strength welds, the MPC enclosure vessel welds provide a secure barrier against leakage. The MPC enclosure vessel is designed to have no credible leakage under all normal, off-normal, and hypothetical accident conditions of storage.

The confinement boundary is defined by the MPC shell, baseplate, MPC lid, port cover plates, closure ring, and associated welds. MPC shell welds and shell to baseplate weld are subject to helium leakage testing. Most confinement boundary welds are inspected by radiography or ultrasonic examination. Field welds are examined by the liquid penetrant method on the root (if more than one weld pass is required) and final passes. In addition to liquid penetrant examination, the MPC lid-to-shell weld is pressure tested, and volumetrically examined or multi-pass liquid penetrant examined. The vent and drain port cover plates are subject to the liquid penetrant examination and helium leakage testing. These inspection and testing techniques are performed to verify the integrity of the confinement boundary.

The helium retention boundary is defined by the overpack baseplate, inner shell, top flange, vent and drain port plugs, and bolted closure plate containing two concentric seals. All welds that form a part of the helium retention boundary are examined by radiography. The overpack welds and seals are helium leakage tested during fabrication to verify their integrity. Helium leakage tests of all overpack closure seals are performed following each loading sequence.

2.2.2.5 Malfunction of Forced Helium Dehydrator (FHD)

The FHD system is a forced helium circulation device used to effectuate moisture removal from loaded MPCs. For circulating helium, the FHD system is equipped with active components requiring external power for normal operation.

Initiating events of FHD malfunction are: (i) a loss of external power to the FHD System and (ii) an active component trip. In both cases a stoppage of forced helium circulation occurs and heat dissipation in the MPC transitions to natural convection cooling.

Although the FHD System is monitored during its operation, stoppage of FHD operations does not require actions to restore forced cooling for adequate heat dissipation. This is because the condition of natural convection cooling evaluated in Section 4.5 shows that the fuel temperatures remain below off-normal limits. An FHD malfunction is detected by operator response to control panel visual displays and alarms.

2.2.3 Environmental Phenomena and Accident Condition Design Criteria

Environmental phenomena and accident condition design criteria are defined in the following subsections.

The minimum acceptance criteria for the evaluation of the accident condition design criteria are that the MPC confinement boundary maintains radioactive material confinement, the MPC fuel basket structure maintains the fuel contents subcritical, and the stored SNF can be retrieved by normal means.

A discussion of the effects of each environmental phenomena and accident condition is provided in Section 11.2. The consequences of each accident or environmental phenomena are evaluated against the requirements of 10CFR72.106 and 10CFR20. Section 11.2 also provides the corrective action for each event. The location of the detailed analysis for each event is referenced in Section 11.2.

2.2.3.1 Handling Accident

The HI-STAR 100 System must withstand loads due to a handling accident. Even though the loaded HI-STAR 100 System will be handled in accordance with approved, written procedures and will use lifting equipment which complies with ANSI N14.6.

If the lifting device complies with the single failure proof criteria specified by the USNRC and cited in Chapter 1, the drop cases specified in Table 2.0.1 and 2.0.2 do not apply. In cases where single failure proof criteria are not used, acceptance criteria for site specific analysis shall meet Table 2.0.1 and 2.0.2.

2.2.3.2 Tip-Over

The HI-STAR 100 System is demonstrated to remain kinematically stable under the design basis environmental phenomena (tornado, earthquake, etc.). However, the cask must also withstand impact due to a postulated tip-over event. The structural integrity of a loaded HI-STAR 100 System after a tip-over onto a reinforced concrete pad is demonstrated using a side drop bounding *analysis as described in Chapter 3*. The cask tip-over is not postulated as an outcome of any environmental phenomenon or accident condition. The cask tip-over is a non-mechanistic event.

During original licensing for the HI-STAR 100 System, a single set of ISFSI pad and subgrade design parameters (now labeled Set A) was established. Experience has shown that achieving the maximum concrete compressive strength (at 28 days) shown in Table 2.2.9 for Set A can be difficult. Therefore, a second set of ISFSI pad and subgrade design parameters (labeled Set B) has been developed. The Set B ISFSI parameters include a thinner concrete pad and less stiff subgrade, which allow for a higher concrete compressive strength. Cask deceleration values for all design basis drop and tipover events have been verified to be less than or equal to the design limit of 60 g's at the top of the fuel basket for both sets of ISFSI pad and subgrade design parameters.

The original set and the new set (Set B) of acceptable ISFSI pad and subgrade design parameters are specified in Table 2.2.9. Users may design their ISFSI pads and subgrade in compliance with either parameter Set A or Set B. Alternatively, users may design their site-specific ISFSI pad and subgrade using any combination of design parameters that result in a structurally competent pad that meets the provisions of ACI 318 and also limits the deceleration of the cask to less than or equal to 60 g's for the design basis drop and tipover events. The structural analyses for site-specific ISFSI pad design shall be performed using methodologies consistent with those described in this FSAR, as applicable.

A tip-over analysis of a loaded HI-STAR 100 overpack stored on an ISFSI pad in the horizontal orientation is not required because:

i) the HI-STAR 100 is analyzed in Appendix 3.A for a 72" horizontal side drop onto an ISFSI pad that complies with the design parameters in Table 2.2.9;

ii) if the HI-STAR 100 overpack is stored horizontally the center of gravity height of the overpack above the ISFSI pad must not exceed 72 inches.

2.2.3.3 Fire

The possibility of a fire accident near an ISFSI site is considered to be extremely remote due to the absence of significant combustible materials. The only credible concern is related to a transport vehicle fuel tank fire engulfing the loaded cask while it is being moved to the ISFSI.

The HI-STAR 100 System must withstand temperatures due to a fire event. The fire accident for storage is conservatively specified to be the result of the spillage and ignition of 50 gallons (190 liters) of combustible transporter fuel. The HI-STAR overpack surfaces are considered to receive an incident radiation and convection heat flux from the fire. Table 2.2.8 provides the fire duration based on the amount of flammable materials assumed. The temperature of the fire is assumed to be 1475 ^{o}F (800 ^{o}C) in accordance with 10CFR71.73.

The accident condition design temperatures for the HI-STAR 100 System, and the fuel rod cladding limits are specified in Table 2.2.3. The specified fuel cladding temperature limits are based on the temperature limits specified in ISG-11, Rev 3 [2.0.5].

2.2.3.4 Partial Blockage of MPC Basket Vent Holes

The HI-STAR 100 System is designed to withstand reduction of flow area due to partial blockage of the MPC basket vent holes. As the MPC basket vent holes are internal to the confinement barrier, the only events that could partially block the vents are fuel cladding failure and debris associated with this failure, or the collection of crud at the base of the stored SNF assembly. The HI-STAR 100 System maintains the SNF in an inert environment with fuel rod cladding temperatures below accepted values (Table 2.2.3). Therefore, there is no credible mechanism for gross fuel cladding degradation during storage in the HI-STAR 100 System. For the storage of damaged BWR fuel assemblies or fuel debris, the assemblies and fuel debris will be placed in damaged fuel containers. The damaged fuel container is equipped with fine mesh screens which ensure that the damaged fuel and fuel debris will not be escape to block the MPC basket vent holes. In addition, each MPC will be loaded once for long-term storage and, therefore, buildup of crud in the MPC due to numerous loadings is precluded. Using crud quantities reported in an Empire State Electric Energy Research Corporation Report [2.2.4], a layer of crud of conservative depth is assumed to partially block the MPC basket vent holes. The crud depths for the different MPCs are listed in Table 2.2.8.

2.2.3.5 Tornado

The HI-STAR 100 System must withstand pressures, wind loads, and missiles generated by a tornado. The prescribed design basis tornado and wind loads for the HI-STAR 100 System are consistent with NRC Regulatory Guide 1.76 [2.2.5], ANSI 57.9 [2.2.6], and ASCE 7-88 [2.2.3]. Table 2.2.4 provides the wind speeds and pressure drop which the HI-STAR 100 System must withstand while maintaining kinematic stability. The small pressure drop is bounded by the accident condition overpack internal design pressure.

The kinematic stability of the HI-STAR 100 System must be demonstrated under impact from tornado-generated missiles in conjunction with the wind loadings. Standard Review Plan (SRP) 3.5.1.4 of NUREG-0800 [2.2.7] stipulates that the postulated missiles include at least three objects: a massive high kinetic energy missile which deforms on impact (large missile), a rigid missile to test penetration resistance (penetrant missile), and a small rigid missile of a size sufficient to pass through any openings in the protective barriers (micro-missile). SRP 3.5.1.4 suggests an automobile for a large missile, a rigid solid steel cylinder for the penetrant missile, and a solid sphere for the small rigid missile, all impacting at 35% of the maximum horizontal wind speed of the design basis tornado. Table 2.2.5 provides the missile data used in the analysis, which is based on the above SRP guidelines. The effects of a tornado missile are considered to bound the effects of a light general aviation airplane crashing on an ISFSI facility.

2.2.3.6 Flood

The HI-STAR 100 System must withstand pressure and water forces associated with a flood. Resultant loads on the HI-STAR 100 System consist of buoyancy effects, static pressure loads, and pressure due to water velocity. The flood is assumed to deeply submerge the HI-STAR 100 System (see Table 2.2.8). The flood water depth is based on the submergence requirement of 10CFR71. This condition corresponds to a hydrostatic pressure which is bounded by the overpack external pressure stated in Table 2.2.1.

It must be shown that the overpack does not collapse, buckle, or allow water in-leakage under the hydrostatic pressure from the flood.

The flood water is assumed to be non-stagnant. The maximum allowable flood water velocity is determined by calculating the equivalent pressure loading required to slide or tip over the HI-STAR 100 System. The design basis flood water velocity is stated in Table 2.2.8. Site-specific safety reviews performed by the licensee must confirm that flood parameters do not exceed the flood depth, slide, or tip-over forces.

Most reactor sites are hydrologically characterized as required by Paragraph 100.10(c) of 10CFR100 [2.2.8] and further articulated in Reg. Guide 1.59, "Design Basis Floods for Nuclear Power Plants" [2.2.9] and Reg. Guide 1.102, "Flood Protection for Nuclear Power Plants" [2.2.10]. It is assumed that a complete characterization of the ISFSI's hydrosphere including the effects of hurricanes, floods, seiches and tsunamis is available to enable a site-specific evaluation of the HI-STAR 100 System for kinematic stability. An evaluation for tsunamis† for certain coastal sites should also be performed to demonstrate that sliding or tip-over will not occur and that the maximum flood depth will not be exceeded.

Analysis for each site for such transient hydrological loadings must be made for that site. It is expected that the plant licensee will perform this evaluation under the provisions of 10CFR Part 72.212.

2.2.3.7 Seismic Design Loadings

† A tsunami is an ocean wave from seismic or volcanic activity or from submarine landslides. A tsunami may be the result of nearby or distant events. A tsunami loading may exist in combination with wave splash and spray, storm surge and tides.

The HI-STAR 100 must withstand loads arising due to a seismic event and must be shown not to tip over during a seismic event. Subsection 3.4.7 contains calculations based on conservative static "incipient tipping" calculations which demonstrate static stability. The calculations in Subsection 3.4.7 result in the value specified in Table 2.2.8, which provide the maximum horizontal zero period acceleration (ZPA) versus vertical acceleration multiplier above which static incipient tipping would occur. This conservatively assumes the peak acceleration values of each of the two horizontal earthquake components and the vertical component occur simultaneously. The maximum horizontal ZPA provided in Table 2.2.8 is the vector sum of two horizontal earthquakes.

2.2.3.8 <u>100% Fuel Rod Rupture</u>

The fuel rod rupture releases shall be per Table 2.0.1

2.2.3.9 <u>Confinement Boundary Leakage</u>

No credible scenario has been identified that would cause failure of the confinement system. *Section 7.1 provides a discussion as to why leakage of any magnitude from the MPC is not credible, based on the materials and methods of fabrication and inspection.*

2.2.3.10 <u>Explosion</u>

The HI-STAR 100 System must withstand loads due to an explosion. The accident condition overpack external pressure specified in Table 2.2.1 bounds all credible external explosion events. There are no credible internal explosion events since all materials are compatible with the various operating environments, as discussed in Subsection 3.4.1. The MPC is composed of stainless steel, neutron absorber material, and aluminum alloy 1100, all of which have a long proven history of use in fuel pool at nuclear power plants. For these materials, and considering the protective measures taken during loading and unloading operations, there is no credible cause for an internal explosive event.

2.2.3.11 <u>Lightning</u>

The HI-STAR 100 System must withstand loads due to lightning. The effect of lightning on the HI-STAR 100 System is evaluated in Chapter 11.

2.2.3.12 <u>Burial Under Debris</u>

The HI-STAR 100 System must withstand burial under debris. Such debris may result from floods, wind storms, or mud slides. The thermal effects of burial under debris on the HI-STAR 100 System is evaluated in Chapter 11. Siting of the ISFSI pad shall ensure that the storage location is not located near shifting soil. Burial under debris is a highly unlikely accident, but is analyzed in this FSAR.

2.2.3.13 <u>Extreme Environmental Temperature</u>

The HI-STAR 100 System must withstand extreme environmental temperatures. The extreme accident level temperature is specified in Table 2.2.2. The extreme accident level temperature occurs with steady-state insolation. The environmental temperature is assumed to persist for a duration sufficient to allow the system to reach steady-state temperatures. The HI-STAR 100 System has a large thermal inertia. Therefore, this temperature is assumed to persist over three days (3-day average).

2.2.4 Applicability of Governing Documents

The ASME Boiler and Pressure Vessel Code (ASME Code), 1995 Edition with Addenda through 1997, is the governing code for the structural design of the HI-STAR 100 System. The ASME Code is applied to each component consistent with the function of the component. Table 2.2.6 lists each structure, system and component (SSC) of the HI-STAR 100 System which are labeled Important to Safety, along with its function and governing Code. Some components perform multiple functions and in those cases, the most restrictive Code is applied. In accordance with NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components" [2.2.11] and according to importance to safety, components of the HI-STAR 100 System are classified as A, B, C, or NITS (not important to safety) in Table 2.2.6 according to NUREG/CR-6407 [2.2.11].

Table 2.2.7 lists the applicable ASME Code section and paragraph for material procurement, design, fabrication and inspection of the components of the HI-STAR 100 System that are governed by the ASME Code. The ASME Code section listed in the design column is the section used to define allowable stresses for structural analyses.

Table 2.2.15 lists the alternatives to the ASME Code for the HI-STAR 100 System and the justification for those alternatives.

Proposed alternatives to the ASME Code, 1995 Edition with Addenda through 1997 including modifications to the alternatives may be used on a case-specific basis when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternative should demonstrate that: The proposed alternatives would provide an acceptable level of quality and safety, or compliance with the specified requirements of the ASME Code, 1995 Edition with Addenda through 1997, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Requests for alternatives shall be submitted in accordance with 10 CFR 72.4.

2.2.5 Service Limits

In the ASME Code, plant and system operating conditions are commonly referred to as normal, upset, emergency, and faulted. Consistent with the terminology in NRC documents, this FSAR utilizes the terms normal, off-normal, and accident conditions.

The ASME Code defines four service conditions in addition to the Design Limits for nuclear components. They are referred to as Level A, Level B, Level C, and Level D service limits, respectively. Their definitions are provided in Paragraph NCA-2142.4 of the ASME Code. The four levels are used in this FSAR as follows:

a. Level A Service Limits: Level A Service Limits are used *to establish allowables* for normal condition load combinations.

b. Level B Service Limits: Level B Service Limits are used to establish *allowables* for off-normal condition load combinations.

c. Level C Service Limits: Level C Service Limits are not used.

d. Level D Service Limits: Level D Service Limits are used to establish *allowables* for accident condition load combinations.

The ASME Code service limits are used in the structural analyses for definition of allowable stresses and allowable stress intensities. Allowable stresses and stress intensities for structural analyses are tabulated in Chapter 3. These service limits are matched with normal, off-normal, and accident condition loads combinations in the following subsections.

The MPC confinement boundary and the overpack helium retention boundary are required to meet Section III, Class 1 stress intensity limits. Table 2.2.10 lists the stress intensity limits for the Levels A, B, and D service limits for Class 1 structures extracted from the ASME Code (1995 Edition). The limits for the MPC fuel basket, required to meet the stress intensity limits of Subsection NG of the ASME Code, are listed in Table 2.2.11. Table 2.2.12 lists allowable stress limits for the external steel structures (intermediate shells, radial channels, and outer enclosure) which are analyzed to meet the stress limits of Subsection NF, Class 3. *Only service levels A, B, and D requirement for normal, off-normal, and accident conditions, are applicable.*

2.2.6 Loads

Subsections 2.2.1, 2.2.2, and 2.2.3 describe the design criteria for normal, off-normal, and accident conditions, respectively. The individual loads listed in Table 2.2.13 are defined from the design criteria. Each load is assigned a symbol for subsequent use in the load combinations.

The loadings listed in Table 2.2.13 fall into two broad categories; namely, (i) those which affect kinematic stability, and (ii) those which produce significant stresses. The loadings in the former category are principally applicable to the overpack. Tornado wind (W'), earthquake (E), and tornado-borne missile (M) are essentially loadings which can destabilize a cask. Analyses reported in Chapter 3 show that the HI-STAR 100 overpack structure will remain kinematically stable under these loadings. *Additionally, for the missile impact case (M), analyses that demonstrate that the overpack structure remains unbreached by the postulated missiles are provided in Chapter 3.*

Loadings in the second category produce global stresses which must be shown to comply with the stress intensity or stress limits, as applicable. The relevant loading combinations for the fuel basket, the MPC, and the overpack are different because of differences in their function.

2.2.7 Load Combinations

To demonstrate compliance with the design requirements for normal, off-normal, and accident conditions of storage, the individual loads, identified in Table 2.2.13, are combined into load combinations. In the formation of the load combinations, it is recognized that the number of combinations requiring detailed analyses is reduced by defining bounding loads. Analyses performed using bounding loads serve to satisfy the requirements for analysis of a multitude of separately identified loads in combination.

The number of loading combinations is reduced by defining the internal and external pressures (P_i and P_o) such that they bound other surface-intensive loads, namely snow (S), tornado wind (W′), flood (F), and explosion (E^*). *Thus, evaluation of pressure in a load combination established for a given storage condition enables many individual load effects to be included in a single load combination.*

Table 2.2.14 *identifies the combinations of the loads that are required to be considered in order to ensure compliance with the design criteria set forth in this chapter. Table 2.2.14 presents the load combinations in terms of the loads that must be considered together. A number of load combinations are established for each ASME Service Level. Within each loading case, there may be more than one analysis that is required to demonstrate compliance. Since the breakdown into specific analyses is most applicable to the structural evaluation, the identification of individual analyses with the applicable loads for each load combination is found in Chapter 3.*

2.2.8 Allowable Stresses

The stress intensity limits for the MPC confinement boundary and the overpack helium retention boundary for the design condition and the service conditions are provided in Table 2.2.10. The stress intensity limits for the MPC fuel basket are presented in Table 2.2.11 (governed by Subsection NG of Section III). The external structures in the overpack meet the stress limits of Subsection NF of ASME Code, Section III for plate and shell components. Limits for the Level D condition are obtained from Appendix F of ASME Code, Section III for the steel structure of the overpack. The MPC confinement boundary stress intensity limits are obtained from ASME Code, Section III, Subsection NB.

The following definitions of terms apply to the tables on stress intensity limits; these definitions are the same as those used throughout the ASME Code:

S_m: Value of Design Stress Intensity listed in ASME Code Section II, Part D, Tables 2A, 2B and 4

S_y: Minimum yield strength at temperature

S_u: Minimum ultimate strength at temperature

The overpack closure bolts are designed in accordance with NUREG/CR-6007, "Stress Analysis of Closure Bolts for Shipping Casks" [2.2.12]. The overpack lifting trunnions and the assorted lifting bolts are designed according to NUREG-0612 [2.2.13] requirements. Table 2.2.16 provides the allowable stress criteria for the closure bolts, lifting trunnions, and lifting eye bolts.

Table 2.2.1

DESIGN PRESSURES

Pressure Location	**Condition**	**Pressure - psig (kPa)**
MPC Internal Pressure	Normal	100 (689)
	Off-Normal	100 (689)
	Accident	200 (1379)
MPC External Pressure/Overpack Internal Pressure	Normal	40 (275)
	Off-Normal	40 (275)
	Accident	60 (414)
Overpack External Pressure	Normal	(0) Ambient
	Off-Normal	(0) Ambient
	Accident	300 (2000)
Overpack Neutron Shield Enclosure Internal Pressure	Normal	30 (207)
	Off-Normal	30 (207)
	Accident	N/A†

† The overpack neutron shield enclosure is equipped with two rupture disks which are set a relief pressure of 30 psig (200 kPa). Therefore, the pressure cannot exceed 30 psig (200 kPa).

Table 2.2.2

ENVIRONMENTAL TEMPERATURES

Condition	**Temperature - ºF (ºC)**	**Comments**
Normal (Bounding Annual Average)	80 (27)	
Off-Normal (3-Day Average)	-40 and 100 (-40 and 38)	-40 ºF (-40 ºC) with no insolation 100 ºF (38 ºC) with insolation
Extreme Accident Level (3-Day Average)	125 (52)	125 ºF (52 ºC) with maximum insolation

Table 2.2.3

DESIGN TEMPERATURES

HI-STAR 100 Component	Normal Condition Design Temp. (Long-Term Events) °F (°C)	Off-Normal and Accident Condition Design Temp. Limits (Short-Term Events) °F (°C)
MPC shell	450 (230)	775 (410)
MPC basket	725 (385)	950 (510)
MPC *neutron absorber*	800 (425)	950(510)
MPC lid	550 (285)	775 (410)
MPC closure ring	400 (200)	775 (410)
MPC baseplate	400 (200)	775 (410)
MPC heat conduction elements	725 (385)	950 (510)
Overpack inner shell	400 (200)	500 (260)
Overpack bottom plate	350 (175)	700 (370)
Overpack closure plate	400 (200)	700 (370)
Overpack top flange	400 (200)	700 (370)
Overpack closure plate seals	400 (200)	1200 (645)
Overpack closure plate bolts	350 (175)	1000 (535)
Overpack port plug seals (vent and drain)	400 (200)	1200 (645)
Overpack port cover seals (vent and drain)	400 (200)	1200 (645)
Neutron shielding	300 (145)	300 (145)
Overpack Neutron Shield Enclosure Shell	300 (145)	1350 (730)
Remainder of overpack	350 (175)	1000 (535)
Fuel Cladding	*752 (400) (Storage)* *752 or 1058 (400 or 570) (Short Term Operations)*[††]	*1058 (570) (Off-Normal and Accident Conditions)*

[††] *Short term operations include MPC drying and onsite transport. The 1058°F (570°C) temperature limit applies to MPCs containing all moderate burnup fuel. The limit for MPCs containing one or more high burnup fuel assemblies is 752°F (400°C).*

Table 2.2.4

TORNADO CHARACTERISTICS

Condition	**Value**
Rotational wind speed - mph (km/h)	290 (470)
Translational speed - mph (km/h)	70 (110)
Maximum wind speed - mph (km/h)	360 (580)
Pressure drop – psi (kPa)	3.0 (205)

Table 2.2.5

TORNADO-GENERATED MISSILES

Missile Description	**Mass (kg)**	**Velocity – mph (km/h)**
Automobile	1800	126 (200)
Rigid solid steel cylinder (8 in (200 mm) diameter)	125	126 (200)
Solid sphere (1 in (25 mm) diameter)	0.22	126 (200)

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STAR 100 SYSTEM

MPC [1,2]

Primary Function	Component [3]	Safety Class [4]	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/ Coatin g	Contact Matl. (if dissimilar)
Confinement	Shell	A	ASME Section III; Subsection NB	Alloy X[5]	See Appendix 1.A	NA	NA
Confinement	Baseplate	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Confinement	Lid (One-piece design and top portion of optional two-piece design)	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A and Table 3.3.1	NA	NA
Confinement	Closure Ring	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Confinement	Port Cover Plates	A	ASME Section III; Subsection NB	Alloy X	See Appendix 1.A	NA	NA
Criticality Control	Basket Cell Plates	A	ASME Section III; Subsection NG	Alloy X	See Appendix 1.A	NA	NA
Criticality Control	*Neutron Absorber*	A	Non-code	NA	NA	NA	*Aluminum/SS*
Shielding	Drain and Vent Shield Block	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Shielding	Plugs for Drilled Holes	NITS	Non-code	*SA 193B8 (or equivalent)*	Per ASME Section II	NA	NA
Shielding	Bottom portion of optional two-piece MPC lid design	B	Non-code	*Alloy X or Carbon Steel*	*See Appendix 1.A, Table 3.3.5 for Carbon Steel*	*Stainless Steel coating when using Carbon Steel*	*Stainless Steel when using Carbon Steel*

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STAR 100 SYSTEM

MPC [1,2]

Primary Function	Component [3]	Safety Class [4]	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/ Coating	Contact Matl. (if dissimilar)
Heat Transfer	Heat Conduction Elements	B	Non-code	Aluminum; Alloy 1100	NA	Sandblast Specified Surfaces	Aluminum/SS
Structural Integrity	Upper Fuel Spacer Column	B	ASME Section III; Subsection NG (only for stress analysis)	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Sheathing	A	Non-code	Alloy X	See Appendix 1.A	Aluminum/SS	NA
Structural Integrity	Shims	NITS	Non-code (shims welded directly to angle *plate or parallel plate* basket supports are ASME Section II)	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Basket Supports (Angled Plate *or Parallel Plates with connecting end shims)*	A	ASME Section III; Subsection NG	Alloy X	See Appendix 1.A	NA	NA
Structural Form	Basket Supports (Flat Plates)	NITS	Non-Code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lift Lug	C	NUREG-0612	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lift Lug Baseplate	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Upper Fuel Spacer Bolt	NITS	Non-code	A193-B8 *(or equiv.)*	Per ASME Section II	NA	NA

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STAR 100 SYSTEM

MPC (1,2)

Primary Function	Component (3)	Safety Class (4)	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/ Coating	Contact Matl. (if dissimilar)
Structural Integrity	Upper Fuel Spacer End Plate	B	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Lower Fuel Spacer Column	B	ASME Section III; Subsection NG (only for stress analysis)	S/S. *See Note 6*	See Appendix 1.A	NA	NA
Structural Integrity	Lower Fuel Spacer End Plate	B	Non-code	Alloy X	See Appendix 1.A	NA	NA
Structural Integrity	Vent Shield Block Spacer	C	Non-code	Alloy X	See Appendix 1.A	NA	NA
Operations	Vent and Drain Tube	C	Non-code	S/S	Per ASME Section II	Thread area surface hardened	NA
Operations	Vent & Drain Cap	C	Non-code	S/S	Per ASME Section II	NA	NA
Operations	Vent & Drain Cap Seal Washer	NITS	Non-code	Aluminum	NA	NA	Aluminum/SS
Operations	Vent & Drain Cap Seal Washer Bolt	NITS	Non-code	Aluminum	NA	NA	NA
Operations	Reducer	NITS	Non-code	Alloy X	See Appendix 1.A	NA	NA
Operations	Drain Line	NITS	Non-code	Alloy X	See Appendix 1.A	NA	NA
Operations	Damaged Fuel Container	C	ASME Section III; Subsection NG	*S/S (Primarily 304 S/S)*	See Appendix 1.A	NA	NA
Operations	Drain Line Guide Tube	NITS	Non-code	S/S	NA	NA	NA

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STAR 100 SYSTEM

MPC [1,2]

Primary Function	Component [3]	Safety Class [4]	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/ Coating	Contact Matl. (if dissimilar)
Operations	*Vent and Drain Tube, Optional*	*C*	*Non-code*	*S/S*	*Per ASME Section II*	*Thread area surface hardened*	*NA*
Operations	*Threaded Disc, Plug Adjustment*	*C*	*Non-code*	*S/S*	*Per ASME Section II*	*NA*	*NA*
Operations	*Vent and Drain Plug*	*C*	*Non-code*	*Aluminum*	*NA*	*NA*	*NA*
Operations	*Thread Shield Cap*	*NITS*	*Non-code*	*Aluminum*	*NA*	*NA*	*NA*
Operations	*Retaining Ring*	*NITS*	*Non-code*	*S/S*	*NA*	*NA*	*NA*

Notes: *1) There are no known residuals on finished component surfaces*
2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
3) Component nomenclature taken from Bill of Materials in Chapter 1.
4) A, B, and C denote important to safety classifications as described in the Holtec QA Program. NITS stands for Not Important to Safety.
5) For details on Alloy X material, see Appendix 1.A.
6) Must be Type 304, 304LN, 316, or 316 LN with tensile strength > 75 ksi (517 MPa), yield strength > 30 ksi (207 MPa) and chemical properties per ASTM A554.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STAR 100 SYSTEM

OVERPACK [1,2]

Primary Function	Component [3]	Safety Class [4]	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Helium Retention	Inner Shell	A	ASME Section III; Subsection NB	SA203-E *or SA350-LF3*	Table 3.3.3	Paint inside surface with Thermaline 450 *(Note 5). External surfaces to be coated with a surface preservative.*	NA
Helium Retention	Bottom Plate	A	ASME Section III; Subsection NB	SA350-LF3	Table 3.3.3	Paint inside surface with Thermaline 450 *(Note 5).*	NA
Helium Retention	Top Flange	A	ASME Section III; Subsection NB	SA350-LF3	Table 3.3.3	Paint inside surface with Thermaline 450. Paint outside surface with Carboline 890 *(Note 5).*	NA
Helium Retention	Closure Plate	A	ASME Section III; Subsection NB	SA350-LF3	Table 3.3.3	Paint inside surface with Thermaline 450. Paint outside surface with Carboline 890 *(Note 5).*	NA
Helium Retention	Closure Plate Bolts	A	ASME Section III; Subsection NB	SB637-N07718	Table 3.3.4	NA	NA
Helium Retention	Port Plug	A	Non-code	SA193-B8	Not required	NA	NA
Helium Retention	Port Plug Seal	A	Non-code	Alloy X750	Not required	NA	NA
Helium Retention	Closure Plate Seal	A	Non-code	*Alloy X750*	Not required	NA	NA
Helium Retention	Port Cover Seal	B	Non-code	Alloy X750	Not required	NA	NA
Shielding	Intermediate Shells	B	ASME Section III; Subsection NF	SA516-70	Table 3.3.2	*Internal surfaces to be coated with a silicone encapsulant (Dow- Corning SYLGARD 567 or equivalent)* Exposed areas of fifth intermediate shell to be painted with Carboline 890 *(Note 5).*	NA
Shielding	Neutron Shield	B	Non-code	Holtite-A	Not required	NA	Holtite/CS
Shielding	Plugs for Drilled Holes	NITS	Non-code	SA193-B7	Not required	NA	NA
Shielding	Removable Shear Ring	B	ASME Section III; Subsection NF	*SA203-E or Carbon*	Table 3.3.3	Paint external surface with Carboline 890.	NA

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STAR 100 SYSTEM

OVERPACK [1,2]

Primary Function	Component [3]	Safety Class [4]	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
				Steel SA 350 LF			
Shielding	Pocket Trunnion Plug Plate	C	Non-code	SA240-304	Not required	NA	NA
Heat Transfer	Radial Channels	B	ASME Section III; Subsection NF	SA515-70	Table 3.3.2	Paint outside surface with Carboline 890 *(Note 5)*.	NA
Rotation Pivot and Shielding	Pocket Trunnion	B	Non-Code	SA705-630, 17-4 pH OR SA564-630, 17-4 pH	Table 3.3.4	NA	NA
Structural Integrity	Lifting Trunnion	A	ANSI N14.6	SB637-N07718	Table 3.3.4	NA	NA
Structural Integrity	Rupture Disk*(Relief Device)*	C	Non-code	Commercial	Not required	NA	Brass-C/S
Structural Integrity	Rupture Disk Plate *(Relief Device Plate)*	C	Non-code	A569 or SA516 Gr. 70	Not required	NA	NA
Structural Integrity	Removable Shear Ring Bolt	C	Non-code	SA193-B7	Not required	NA	NA
Structural Integrity	Thermal Expansion Foam	NITS	Non-code	Silicone Foam	Not required	NA	Silicone with CS, brass, and Holtite
Structural Integrity	Closure Bolt Washer	NITS	Non-code	ASTM 564, 17-7 pH	Not required	NA	NA
Structural Integrity	Enclosure Shell Panels	B	ASME Section III; Subsection NF	SA515-70	Table 3.3.2	Paint outside surface with Carboline 890 *(Note 5)*.	NA
Structural Integrity	Enclosure Shell Return	B	ASME Section III; Subsection NF	SA515-70	Table 3.3.2	Paint outside surface with Carboline 890 *(Note 5)*.	NA
Structural Integrity	Port Cover	B	ASME Section III; Subsection NF	SA203E or	Table 3.3.3	Paint outside surface with Carboline 890 *(Note 5)*.	NA

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STAR 100 SYSTEM

OVERPACK (1,2)

Primary Function	Component (3)	Safety Class (4)	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
				SA 350 LF			
Structural Integrity	Port Cover Bolt	C	Non-code	SA193-B7	Not required	NA	NA
Operations	Trunnion Locking Pad and End Cap Bolt	C	Non-code	SA193-B7	Not required	NA	NA
Operations	Lifting Trunnion End Cap	C	Non-code	SA516-70 or SA515 Gr. 70	Table 3.3.2	Paint exposed surfaces with Carboline 890. *(Note 5)*	NA
Operations	Lifting Trunnion Locking Pad	C	Non-code	SA516-70	Table 3.3.2	Paint exposed surfaces with Carboline 890 *(Note 5)*.	NA
Operations	Nameplate	NITS	Non-code	S/S	Not required	NA	NA

Notes: 1) There are no known residuals on finished component surfaces.
2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III. For parts beyond the purview of ASME Section III, compliance with Section IX and Section II of the Code shall be observed to the extent practicable.
3) Component nomenclature taken from drawings in Chapter 1.
4) A, B and C denote important to safety classifications as described in NUREG/CR-6407. NITS stands for Not Important to Safety.
5) Thermaline 450 and Carboline 890 were the product names at the time of initial licensing. Chemically identical products with different names are permitted. For example, Carboline 890 was re-named Carboguard 890 in 2000, with no change to the coating material and is, therefore, acceptable for use where Carboline 890 is specified.

Table 2.2.7

HI-STAR 100 ASME BOILER AND PRESSURE VESSEL CODE APPLICABILITY

HI-STAR 100 Component	Material Procurement	Design	Fabrication	Inspection
Overpack helium retention boundary	Section II, Section III, Subsection NB, NB-2000	Section III, Subsection NB, NB-3200	Section III, Subsection NB, NB-4000	Section III, Subsection NB, NB-5000 and Section V
Overpack intermediate shells, radial channels, outer enclosure	Section II, Section III, Subsection NF, NF-2000	Section III, Subsection NF, NF-*3300*	Section III, Subsection NF, NF-4000	Section III, Subsection NF, NF-5360 and Section V
MPC confinement boundary	Section II, Section III, Subsection NB, NB-2000	Section III, Subsection NB, NB-3200	Section III, Subsection NB, NB-4000	Section III, Subsection NB, NB-5000 and Section V
MPC fuel basket	Section II, Section III, Subsection NG, NG-2000 *for core support structures (NG-1121)*	Section III, Subsection NG, NG-3300 and NG-3200 *for core support structures (NG-1121)*	Section III, Subsection NG, NG-4000 *for core support structures (NG-1121)*	Section III, Subsection NG, NG-5000 and Section V *for core support structures (NG-1121)*
Lifting Trunnions	Section II, Section III, Subsection NF, NF-2000	ANSI N14.6	Section III, Subsection NF, NF-4000	*ANSI 14.6* See Chapter 9
MPC basket supports *(Angled Plates)*	Section II, Section III, Subsection NG, NG-2000 *for internal structures (NG-1122)*	Section III, Subsection NG, NG-3300 and NG-3200 *for internal structures (NG-1122)*	Section III, Subsection NG, NG-4000 *for internal structures (NG-1122)*	Section III, Subsection NG, NG-5000 and Section V *for internal structures (NG-1122)*
Damaged fuel Container	Section II, Section III, Subsection NG, NG-2000	Section III, Subsection NG, NG-3300 and NG-3200	Section III, Subsection NG, NG-4000	Section III, Subsection NG, NG-5000 and Section V

Table 2.2.8

ADDITIONAL DESIGN INPUT DATA FOR NORMAL, OFF-NORMAL, AND ACCIDENT CONDITIONS

Item	Condition	Value
Snow Pressure Loading - lb./ft^2 (kg/m^2)	Normal	100 (490)
Constriction of MPC Basket Vent Opening By Crud Settling (Depth of Crud, in. (mm))	Accident	0.85 (20) (MPC-68) 0.36 (9) (MPC-24) 0.36 (9) (MPC-32)
Cask Environment During the Postulated Fire Event - ºF (ºC)	Accident	1475 (800)
Fire Duration (seconds)	Accident	305
Maximum submergence depth due to flood - ft (m)	Accident	656 (200)
Flood water velocity - ft/s (m/s)	Accident	13 (4)
Maximum Horizontal ZPA (Zero Period Acceleration) for vertically arrayed HI-STAR† (g's)	Accident	0.314 (w/1.0 vertical) 0.332 (w/0.75 vertical) 0.339 (w/0.667 vertical) 0.354 (w/0.5 vertical)

† The maximum horizontal ZPA is specified as the vector sum of the g-loading in two orthogonal directions as a function of the vertical acceleration multiplier which is the maximum vertical ZPA divided by the maximum horizontal ZPA for a single orthogonal direction for the site.

Table 2.2.9

EXAMPLES OF ACCEPTABLE ISFSI PAD DESIGN PARAMETERS

PARAMETER	**PARAMETER SET 'A'**†	**PARAMETER SET 'B'**
Concrete thickness, t_p	≤ 36 inches (90 cm)	≤ 28 inches (70 cm)
Concrete Compressive Strength (at 28 days), f_c'	≤ 4,200 psi (29 MPa)	≤ 6,000 psi (40 MPa)
Reinforcement Top and Bottom (both directions)	Reinforcing bar shall be 60 ksi (410 MPa) yield strength ASTM material	Reinforcing bar shall be 60 ksi (410 MPa) yield strength ASTM material
Subgrade Effective Modulus of Elasticity†† (measured prior to ISFSI pad installation), E	≤ 28,000 psi (190 MPa)	≤ 16,000 psi (110 MPa)

NOTE: A static coefficient of friction of 0.53 between the ISFSI pad and the bottom of the overpack shall be used. If for a specific ISFSI a higher value of the coefficient of friction is used, it shall be verified by test. The test procedure shall follow the guidelines included in the Sliding Analysis in Subsection 3.4.7.

† *The characteristics of this pad are identical to the pad considered by Lawrence Livermore Laboratory (see Appendix 3.A).*

†† An acceptable method of defining the soil effective modulus of elasticity applicable to the drop and tip-over analysis is provided in Table 13 of NUREG/CR-6608 with soil classification in accordance with ASTM-D2487 Standard Classification of Soils for Engineering Purposes (Unified Soil Classification System USCS) and density determination in accordance with ASTM-D-1586 Standard Test Method for Penetration Test and Split/barrel Sampling of Soils.

Table 2.2.10
MPC CONFINEMENT BOUNDARY AND OVERPACK HELIUM RETENTION BOUNDARY
STRESS INTENSITY LIMITS
FOR DIFFERENT LOADING CONDITIONS (ELASTIC ANALYSIS PER NB-3220)†

STRESS CATEGORY	**DESIGN**	**LEVELS A & B**	**LEVEL D††**
Primary Membrane, P_m	S_m	N/A†††	AMIN ($2.4S_m$, $.7S_u$)
Local Membrane, P_L	$1.5S_m$	N/A	150% of P_m Limit
Membrane plus Primary Bending	$1.5S_m$	N/A	150% of P_m Limit
Primary Membrane plus Primary Bending	$1.5S_m$	N/A	150% of P_m Limit
Membrane plus Primary Bending plus Secondary	N/A	$3S_m$	N/A
Average Shear Stress††††	$0.6S_m$	$0.6S_m$	$0.42S_u$

† Stress combinations including F (peak stress) apply to fatigue evaluations only.

†† Governed by Appendix F, Paragraph F-1331 of the ASME Code, Section III.

††† No specific stress intensity limit applicable.

†††† Governed by NB-3227.2 or F-1331.1(d)

Table 2.2.11

MPC BASKET STRESS INTENSITY LIMITS
FOR DIFFERENT LOADING CONDITIONS (ELASTIC ANALYSIS PER NG-3220)

STRESS CATEGORY	**DESIGN**	**LEVELS A & B**	**LEVEL D**†
Primary Membrane, P_m	S_m	S_m	AMIN ($2.4S_m$, $.7S_u$)††
Primary Membrane plus Primary Bending	$1.5S_m$	$1.5S_m$	150% of P_m Limit
Primary Membrane plus Primary Bending plus Secondary	N/A†††	$3S_m$	N/A

† Governed by Appendix F, Paragraph F-1331 of the ASME Code, Section III.

†† Governed by NB-3227.2 or F-1331.1(d)

††† No specific stress intensity limit applicable.

Table 2.2.12

STRESS LIMITS FOR DIFFERENT
LOADING CONDITIONS FOR THE EXTERNAL STRUCTURES IN THE HI-STAR OVERPACK
(ELASTIC ANALYSIS PER NF-3260)

	SERVICE CONDITION†		
STRESS CATEGORY	**DESIGN + LEVEL A**	**LEVEL B**	**LEVEL D**††
Primary Membrane, P_m	S	1.33S	AMAX (1.2S_y, 1.5S_m) but < .7S_u
Primary Membrane, P_m, plus Primary Bending, P_b	1.5S	1.995S	150% of P_m
Shear Stress (Average)	0.6S	0.6S	<0.42S_u

Definitions:

S = Allowable Stress Value for Table 1A, ASME Section II, Part D
S_m = Allowable Stress Intensity Value from Table 2A, ASME Section II, Part D
S_u = Ultimate Strength

† Limits for Design and Levels A and B are on maximum stress.
Limits for Level D are on maximum stress intensity.
†† Governed by Appendix F, Paragraph F-1332 of the ASME Code, Section III.

Table 2.2.13
NOTATION FOR DESIGN LOADINGS FOR NORMAL, OFF-NORMAL, AND ACCIDENT CONDITIONS

NORMAL CONDITION	
LOADING	**NOTATION**
Dead Weight	D
Handling Loads	H
Design Pressure (Internal)†	P_i
Design Pressure (External)†	P_o
Snow	S
Wind	W
Operating Temperature	T
OFF-NORMAL CONDITION	
LOADING	**NOTATION**
Off-Normal Pressure (Internal)†	P_i
Off-Normal Pressure (External)†	P_o
Off-Normal Temperature	T′

† Internal Design Pressure P_i bounds the normal and off-normal condition internal pressures. External Design Pressure P_o bounds off-normal external pressures. Similarly, accident pressures P_i^* and P_o^*, respectively, bound actual internal and external pressures under all postulated environment phenomena and accident events.

Table 2.2.13 (continued)

NOTATION FOR DESIGN LOADINGS FOR NORMAL, OFF-NORMAL, AND ACCIDENT CONDITIONS

ACCIDENT CONDITIONS	
LOADING	**NOTATION**
Handling Accident (Drop)	H′
Earthquake	E
Fire	T^*
Tornado Missile	M
Tornado Wind	W′
Flood	F
Explosion	E^*
Accident Pressure (Internal)	P_i^*
Accident Pressure (External)	P_o^*

Table 2.2.14

APPLICABLE LOAD CASES AND COMBINATIONS FOR EACH CONDITION AND COMPONENT[†, ††]

CONDITION	**LOADING CASE**	**MPC**	**OVERPACK**
Design (ASME Code Pressure Compliance)	1	P_i, P_o	P_i, P_o
Normal (Level A)	1	D ,T , H , P_i	D , T , H , P_i
	2	D , T , H , P_o	N/A
Off-Normal (Level B)	1	D , T′, H , P_i	D , T′ , H , P_i
	2	D , T′ , H , P_o	N/A
Accident (Level D)	1	D , T , P_i , H′	D , T , P_i , H′
	2	D, T^*, P_i^*	D , T^*, P_i^*
	3	D, T^*, P_o^*	D , T^* , P_o^*[†††]

† The loading notations are given in Table 2.2.13. Each symbol represents a loading type and may have different values for different components.

†† N/A stands for "Not Applicable".

††† P_o* bounds the external pressure due to explosion.

Table 2.2.15 (continued)

LIST OF ASME CODE ALTERNATIVES FOR HI-STAR 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC, MPC basket assembly, and HI-STAR overpack steel structure.	*Subsection NCA*	*General Requirements. Requires preparation of a Design Specification, Design Report, Overpressure Protection Report, Certification of Construction Report, Data Report, and other administrative controls for an ASME Code stamped vessel.*	*Because the MPC and overpack are not ASME Code stamped vessels, none of the specifications, reports, certificates, or other general requirements specified by NCA are required. The HI-STAR FSAR includes the design criteria, service conditions, and load combinations for the design and operation of the HI-STAR 100 System as well as the results of the stress analyses to demonstrate that applicable Code stress limits are met. Additionally, the fabricator is not required to have an ASME-certified QA program. All important-to-safety activities are governed by the NRC approved Holtec QA program.* *Because the cask components are not certified to the Code, the terms "Certificate Holder" and "Inspector" are not germane to the manufacturing of NRC-certified cask components. To eliminate ambiguity, the responsibilities assigned to the Certificate Holder in the various articles of Subsections NB, NG, and NF of the Code, as applicable, shall be interpreted to apply to the NRC Certificate of Compliance (CoC) holder (and by extension, to the component fabricator) if the requirement must be fulfilled. The Code term "Inspector" means the QA/QC personnel of the CoC holder and its vendors assigned to oversee and inspect the manufacturing process.*
MPC	NB-1100	Statement of requirements for Code stamping of components.	MPC enclosure vessel is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.
MPC	NB-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec approved suppliers with Certified Material Test Reports (CMTRs) in accordance with NB-2000 requirements.
MPC basket	*NB-1130*	*NB-1132.2(d) requires that the*	*The MPC basket supports (nonpressure-retaining structural*

Table 2.2.15 (continued)

LIST OF ASME CODE ALTERNATIVES FOR HI-STAR 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
supports and lift lugs		*first connecting weld of a nonpressure- retaining structural attachment to a component shall be considered part of the component unless the weld is more than 2t from the pressure-retaining portion of the component, where t is the nominal thickness of the pressure-retaining material.* *NB-1132.2(e) requires that the first connecting weld of a welded nonstructural attachment to a component shall conform to NB-4430 if the connecting weld is within 2t from the pressure-retaining portion of the component.*	*attachments) and lift lugs (nonstructural attachments used exclusively for lifting an empty MPC) are welded to the inside of the pressure-retaining MPC shell, but are not designed in accordance with Subsection NB. The basket supports and associated attachment welds are designed to satisfy the stress limits of Subsection NG and the lift lugs and associated attachment welds are designed to satisfy the stress limits of Subsection NF, as a minimum. These attachments and their welds are shown by analysis to meet the respective stress limits for their service conditions.* *Likewise, non-structural items, such as shield plugs, spacers, etc. if used, can be attached to pressure-retaining parts in the same manner.*
MPC, MPC basket assembly, and HI-STAR overpack steel structure.	NB-3100 NG-3100 NF-3100	Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.	These requirements are not applicable. The HI-STAR FSAR, serving as the Design Specification, establishes the service conditions and load combinations for the storage system.
MPC	*NB-3350*	*NB-3352.3 requires, for Category C joints, that the minimum dimensions of the welds and throat thickness shall be as shown in Figure NB-*	*The MPC shell-to-baseplate weld joint design (designated Category C) may not include a reinforcing fillet weld or a bevel in the MPC baseplate, which makes it different than any of the representative configurations depicted in Figure NB-4243-1. The transverse thickness of this weld is equal to the thickness of*

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
		4243-1.	*the adjoining shell (1/2 inch (13 mm)). The weld is designed as a full penetration weld that receives VT and RT or UT, as well as final surface PT examinations. Because the MPC shell design thickness is considerably larger than the minimum thickness required by the Code, a reinforcing fillet weld that would intrude into the MPC cavity space is not included. Not including this fillet weld provides for a higher quality radiographic examination of the full penetration weld.* *From the standpoint of stress analysis, the fillet weld serves to reduce the local bending stress (secondary stress) produced by the gross structural discontinuity defined by the flat plate/shell junction. In the MPC design, the shell and baseplate thicknesses are well beyond that required to meet their respective membrane stress intensity limits.*
MPC, MPC basket assembly, and HISTAR overpack steel structure	*NB-4120* *NG-4120* *NF-4120*	*NB-4121.2, NG-4121.2, and NF-4121.2 provide requirements for repetition of tensile or impact tests for material subjected to heat treatment during fabrication or installation.*	*In-shop operations of short duration that apply heat to a component, such as plasma cutting of plate stock, welding, machining, coating, and pouring of Holtite are not, unless explicitly stated by the Code, defined as heat treatment operations.* *For the steel parts in the HI-STAR 100 System components, the duration for which a part exceeds the off-normal temperature limit shall be limited to 24 hours in a particular manufacturing process (such as the Holtite pouring process).*
MPC and HI-STAR overpack steel structure	*NB-4220* *NF-4220*	*Requires certain forming tolerances to be met for cylindrical, conical, or spherical shells of a vessel.*	*The cylindricity measurements on the rolled shells are not specifically recorded in the shop travelers, as would be the case for a Code-stamped pressure vessel. Rather, the requirements on inter-component clearances (such as the MPC-to-overpack) are guaranteed through fixture-controlled manufacturing. The*

Table 2.2.15 (continued)

LIST OF ASME CODE ALTERNATIVES FOR HI-STAR 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
			fabrication specification and shop procedures ensure that all dimensional design objectives, including inter-component annular clearances are satisfied. *The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components. Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components.*
MPC Lid and Closure Ring Welds	NB-4243	Full penetration welds required for Category C Joints (flat head to main shell per NB-3352.3)	MPC lid and closure ring are not full penetration welds. They are welded independently to provide a redundant seal. Additionally, a weld efficiency factor of 0.45 has been applied to the analyses of these welds.
MPC Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required	Only UT or multi-layer liquid penetrant (PT) examination is permitted. If PT alone is used, at a minimum, it will include the root and final weld layers and each approximately 3/8 inch(9.5 mm) of weld depth.
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Root (if more than one weld pass is required) and final liquid penetrant examination to be performed in accordance with NB-5245. The MPC vent and drain cover plate welds are leak tested. The closure ring provides independent redundant closure for vent and drain cover plates.

Table 2.2.15 (continued)

LIST OF ASME CODE ALTERNATIVES FOR HI-STAR 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	The MPC enclosure vessel is seal welded in the field following fuel assembly loading. The MPC enclosure vessel shall then be pressure tested as defined in Chapter 9. Accessibility for leakage inspections preclude a Code compliant pressure test. All MPC enclosure vessel welds (except the closure ring and vent/drain cover plate welds) are inspected by volumetric examination, except the MPC lid-to shell weld shall be verified by volumetric or multi-layer PT examination. If PT alone is used, at a minimum, it must include the root and final layers and each approximately 3/8 inch (9.5 mm) of weld depth. For either UT or PT, the maximum undetectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded. The vent/drain cover plate weld is confirmed by leakage testing and liquid penetrant examination and the closure ring weld is confirmed by liquid penetrant examination. The inspection process, including findings (indications) shall be made a permanent part of the certificate holder's records by video, photographic, or other means which provide an equivalent retrievable record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME Code Section III, NB-5350 for PT or NB-5332 for UT.

Table 2.2.15 (continued)

LIST OF ASME CODE ALTERNATIVES FOR HI-STAR 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC Enclosure Vessel	NB-7000	Vessels are required to have overpressure protection.	No overpressure protection is provided. Function of MPC enclosure vessel is to contain radioactive contents under normal, off-normal, and accident conditions of storage. MPC vessel is designed to withstand maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.
MPC Enclosure Vessel	NB-8000	States requirements for nameplates, stamping and reports per NCA-8000.	HI-STAR 100 System to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Holtec approved QA program.
Overpack Helium Retention Boundary	NB-1100	Statement of requirements for Code stamping of components.	Overpack helium retention boundary is designed, and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.
Overpack Helium Retention Boundary	NB-2000	Requires materials to be supplied by ASME approved Material Supplier.	Materials will be supplied by Holtec approved suppliers with CMTRs per NB-2000.
Overpack Helium Retention Boundary	NB-7000	Vessels are required to have overpressure protection.	No overpressure protection is provided. Function of overpack vessel is to contain helium contents under normal, off-normal, and accident conditions. Overpack vessel is designed to withstand maximum internal pressure and maximum accident temperatures.
Overpack Helium Retention Boundary	NB-8000	Statement of Requirements for nameplates, stamping and reports per NCA-8000.	HI-STAR 100 System to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. QA data package to be in accordance with Holtec's approved QA program.
MPC Basket Assembly	NG-2000	Requires materials to be supplied by ASME approved Material Supplier.	Materials will be supplied by Holtec approved supplier with CMTRs in accordance with NG-2000 requirements.

Table 2.2.15 (continued)

LIST OF ASME CODE ALTERNATIVES FOR HI-STAR 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC Basket Assembly	*NG-4420*	*NG-4427(a) allows a fillet weld in any single continuous weld to be less than the specified fillet weld dimension by not more than 1/16 inch (1.6 mm), provided that the total undersize portion of the weld does not exceed 10 percent of the length of the weld. Individual undersize weld portions shall not exceed 2 inches (51 mm) in length.*	*Modify the Code requirement (intended for core support structures) with the following text prepared to accord with the geometry and stress analysis imperatives for the fuel basket: For the longitudinal MPC basket fillet welds, the following criteria apply: 1) The specified fillet weld throat dimension must be maintained over at least 92 percent of the total weld length. All regions of undersized weld must be less than 3 inches (75 mm) long and separated from each other by at least 9 inches (230 mm). 2) Areas of undercuts and porosity beyond that allowed by the applicable ASME Code shall not exceed 1/2 inch (13 mm) in weld length. The total length of undercut and porosity over any 1- foot (0.300 m) length shall not exceed 2 inches (50 mm). 3) The total weld length in which items (1) and (2) apply shall not exceed a total of 10 percent of the overall weld length. The limited access of the MPC basket panel longitudinal fillet welds makes it difficult to perform effective repairs of these welds and creates the potential for causing additional damage to the basket assembly (e.g., to the neutron absorber and its sheathing) if repairs are attempted. The acceptance criteria provided in the foregoing have been established to comport with the objectives of the basket design and preserve the margins demonstrated in the supporting stress analysis. From the structural standpoint, the weld acceptance criteria are established to ensure that any departure from the ideal, continuous fillet weld seam would not alter the primary bending stresses on which the design of the fuel baskets is predicated. Stated differently, the permitted weld discontinuities are limited in size to ensure that they remain classifiable as local stress elevators ("peak stress", F, in the ASME Code for which specific stress intensity limits do not apply).*

Table 2.2.15 (continued)

LIST OF ASME CODE ALTERNATIVES FOR HI-STAR 100 SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
MPC Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STAR 100 System will be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. No Code stamping is required. The MPC basket data package will be in conformance with Holtec's QA program.
Overpack Intermediate Shells	NB-4622	All welds, including repair welds, shall be post-weld heat treated (PWHT).	PWHT of intermediate shell-to-top flange and intermediate shell-to-bottom plate welds do not require PWHT. These welds attach non-pressure retaining parts to pressure retaining parts. The pressure retaining parts are > 7 inches (177.8 mm) thick. Localized PWHT will cause material away from the weld to experience elevated temperatures which will have an adverse effect on the material properties.
Overpack Helium Retention Boundary	NG-2000	Perform radiographic examination after post-weld heat treatment (PWHT).	Radiography of the helium retention boundary welds after PWHT is not required. All welds (including repairs) will have passed radiographic examination prior to PWHT of the entire containment boundary. Confirmatory radiographic examination after PWHT is not necessary because PWHT is not known to introduce new weld defects in nickel steels.
Overpack Intermediate Shells	NF-2000	Requires materials to be supplied by ASME approved Material Supplier.	Materials will be supplied by Holtec approved supplier with CMTRs in accordance with NF-2000 requirements.
Overpack Helium Retention Boundary	NB-2330	Defines the methods for determining the T_{NDT} for impact testing of materials.	T_{NDT} shall be defined in accordance with Regulatory Guides 7.11 and 7.12 for the helium retention boundary components.
Overpack Containment	*NF-3320*	*NF-3324.6 and NF-4720*	*These Code requirements are applicable to linear structures*

Table 2.2.15 (continued)

LIST OF ASME CODE ALTERNATIVES FOR HI-STAR 100 SYSTEM

Component	**Reference ASME Code Section/Article**	**Code Requirement**	**Alternative, Justification & Compensatory Measures**
Boundary	*NF-4720*	*provide requirements for bolting.*	*wherein bolted joints carry axial, shear, as well as rotational (torsional) loads. The overpack bolted connections in the structural load path are qualified by design based on the design loadings defined in the FSAR. Bolted joints in these components see no shear or torsional loads under normal storage conditions. Larger clearances between bolts and holes may be necessary to ensure shear interfaces located elsewhere in the structure engage prior to the bolts experiencing shear loadings (which occur only during side impact scenarios).* *Bolted joints that are subject to shear loads in accident conditions are qualified by appropriate stress analysis. Larger bolt-to-hole clearances help ensure more efficient operations in making these bolted connections, thereby minimizing time spent by operations personnel in a radiation area. Additionally, larger bolt-to-hole clearances allow interchangeability of the lids from one particular fabricated cask to another.*
MPC-68 Serial #1021-023, 036, 037 Closure Rings	*NB-2531*	*Requires UT inspection of plate.*	*The sole deviation of the 3/8" (9.5 mm) thick austenitic stainless steel material used for the MPC closure ring is the omission of a straight beam UT inspection as required by NB-2531. The ASME Code required straight beam inspection for vessels because the predominant indication in plates is laminations. Straight beam inspection cannot detect indications perpendicular to the surface of the plate. With respect to maintaining confinement, an indication perpendicular to the surface of the plate is the most critical. Laminations in the plate parallel to the surface of the plate cannot cause leakage through the plate. Therefore, the straight beam UT inspection does not add any value for detecting a defect in the thin closure ring with respect to its confinement function.*

Table 2.2.16

NON-ASME CODE STRESS ALLOWABLE CRITERIA

OVERPACK CLOSURE BOLTS†:

STRESS CATEGORY	NORMAL AND OFF-NORMAL CONDITIONS	ACCIDENT CONDITIONS
Average Tensile Stress	2/3 S_y	AMIN(S_y, 0.7 S_u)
Average Shear Stress	0.6 (2/3 S_y)	AMIN(0.6 S_y, 0.42 S_u)
Combined Tensile and Shear Stress††	$R_t^2 + R_s^2 < 1.0$	$R_t^2 + R_s^2 < 1.0$

LIFTING TRUNNIONS AND LIFTING BOLTS:

The lifting trunnions and the lifting bolts, for the overpack closure plate and for the MPC lid, are designed in accordance with NUREG-0612 and ANSI N14.6. Specifically, the design must meet factors of safety of six based on the material yield stress and ten based on the material ultimate stress for non-redundant lifting devices.

† The overpack closure bolts are designed in accordance with NUREG/CR-6007, "Stress Analysis of Closure Bolts for Shipping Casks".

†† R_t and R_s are the ratios of actual stress to allowable tensile and shear stress, respectively.

Table 2.2.17

INTENTIONALLY DELETED

2.3 SAFETY PROTECTION SYSTEMS

2.3.1 General

The HI-STAR 100 System is a storage and transport cask engineered to provide safe long-term storage of spent nuclear fuel. The HI-STAR 100 System will withstand all normal, off-normal, and postulated accident conditions without any uncontrolled release of radioactive material or excessive radiation exposure to workers or members of the public. Special considerations in the design have been made to ensure long-term integrity and confinement of the stored SNF throughout all cask operating conditions. The design considerations which have been incorporated into the HI-STAR 100 System to ensure safe long-term fuel storage are:

1. The MPC confinement barrier is an enclosure vessel designed in accordance with the ASME Code, Subsection NB with most confinement welds inspected by radiography (RT) or ultrasonic testing (UT). Where RT or UT is not possible, a redundant closure system is provided, which are pressure tested and/or inspected by the liquid penetrant method (See Section 9.1).

2. The MPC confinement barrier is encapsulated by the overpack helium retention boundary.

3. The HI-STAR 100 System is designed to meet the requirements of storage while maintaining the safety of the spent nuclear fuel.

4. The spent nuclear fuel once initially loaded in the MPC and overpack does not require any further canister transfer or fuel handling operations for storage or transport.

5. The decay heat emitted by the spent nuclear fuel is rejected from HI-STAR 100 through passive means. No active cooling systems are employed.

It is recognized that a rugged design with large safety margins is essential, but not sufficient to ensure acceptable performance over the service life of any system. A carefully planned oversight and surveillance plan which does not diminish system integrity but provides reliable information on the effect of passage of time on the performance of the system is essential. Such a surveillance and performance assay program will be developed to be compatible with the specific conditions of the licensee's facility where the HI-STAR 100 System is installed. The general requirements for the acceptance testing and maintenance programs are provided in Chapter 9. Surveillance requirements are specified in Chapter 12.

The structures, systems, and components of the HI-STAR 100 System designated as important to safety are identified in Table 2.2.6. Similar categorization of structures, systems, and components, which are part of the ISFSI, but not part of the HI-STAR 100 System, will be the responsibility of the 10CFR72 licensee.

2.3.2 Protection by Multiple Confinement Barriers and Systems

2.3.2.1 <u>Confinement Barriers and Systems</u>

The radioactivity which the HI-STAR 100 must confine originates from the spent fuel assemblies and, to a lesser extent, the contaminated water in the fuel pool. This radioactivity is confined by multiple confinement barriers.

Radioactivity from the fuel pool water is minimized by preventing contact, removing the contaminated water, and decontamination.

An inflatable seal in the annular gap between the MPC and overpack prevents the fuel pool water from contacting the exterior of the MPC and interior of the overpack while submerged for fuel loading. The fuel pool water is drained from the interior of the MPC, and the MPC internals are dried. The exterior of the overpack has a painted surface which is decontaminated to acceptable levels. Any residual radioactivity deposited by the fuel pool water is confined by the MPC confinement boundary along with the spent nuclear fuel.

The HI-STAR 100 System is designed with several confinement barriers for the radioactive fuel contents. Fuel assemblies classified as damaged fuel or fuel debris are placed within a damaged fuel container which restricts the release of fuel debris. Intact fuel assemblies have cladding which provides the first boundary preventing release of the fission products. The MPC is a seal welded enclosure which provides the confinement boundary. The MPC confinement boundary is defined by the MPC baseplate, shell, lid, closure ring, and port cover plates. An entirely redundant boundary is provided by the overpack helium retention boundary; however, no credit is taken for the overpack helium retention boundary other than its ability to retain a helium atmosphere.

The MPC confinement boundary has been designed to withstand any postulated off-normal operations, internal change, or external natural phenomena. The MPC is designed to endure normal, off-normal, and accident conditions of storage with the maximum decay heat loads without loss of confinement. Designed in accordance with the ASME Code, Section III, Subsection NB, the MPC confinement boundary provides assurance that there will be no release of radioactive materials from the cask under the specified loading conditions. *Redundant closure of the MPC is provided by the MPC closure ring welds which provide a second barrier to the release of radioactive material from the MPC internal cavity.* Therefore, no monitoring system for the confinement boundary is required.

Confinement is discussed further in Chapter 7. MPC field weld examinations, pressure testing, and helium leak testing are performed to verify the confinement function in accordance with the operating procedures in Chapter 8 and the testing and acceptance requirements in Chapter 9.

2.3.2.2 Cask Cooling

To facilitate the passive heat removal capability of the HI-STAR 100, several thermal design criteria are established for normal and off-normal conditions. They are as follows:

- A concentric set of metallic seals in the overpack closure plate contain the helium atmosphere between the exterior of the MPC and the interior of the overpack. A maximum steady-state temperature limit is conservatively set for the seals in Table 2.2.3 based on the manufacturer's specifications. The seals can also withstand the pressures specified in Table 2.2.1.

- The overpack helium retention boundary ensures that the helium atmosphere in the overpack annulus is maintained during normal, off-normal, and accident conditions.

- The MPC confinement boundary ensures that the helium atmosphere inside the MPC is maintained during normal and off-normal conditions. The MPC confinement boundary maintains the helium confinement atmosphere below the design temperatures and pressures stated in Table 2.2.3 and Table 2.2.1, respectively.

- The MPC thermal design maintains the fuel rod cladding temperatures below the values stated in Chapter 4 such that fuel cladding is not degraded during the normal storage period.

- The fabrication method used for layering of the intermediate shells will assure contact between layers. However, the thermal evaluation is conservatively based on small uniform gaps between each shell. During normal conditions, the internal heat source (decay heat from the fuel) acts to provide further assurance of inter-shell contact due to thermal expansion. Likewise, during the fire accident, the outermost intermediate shell would tend to expand more than the inner intermediate shells, possibly reducing the inter-surface contact between them. Such a reduction in contact would tend to insulate the cask during the fire accident. However, no credit is taken for this differential thermal expansion in the thermal analyses. Differential thermal expansion and the resultant stresses caused by restriction of free expansion are, however, accounted for in the structural analyses.

- The heat rejection capacity of the HI-STAR 100 is deliberately understated by conservatively determining the design basis fuel. The decay heat value in Table 2.1.6 is prepared by computing the decay heat from the design basis fuel assembly which produces the highest heat generation rate for a given burnup. Additional margin is built into the calculated cask cooling rate by using a fictitious fuel assembly which offers maximum resistance to the transmission of heat (minimum thermal conductivity).

2.3.3 Protection by Equipment and Instrumentation Selection

2.3.3.1 Equipment

Design criteria for the HI-STAR 100 System is described in Section 2.2. The HI-STAR 100 System may include support equipment. The lifting equipment used to handle the HI-STAR 100 System in and out of the pool is classified as Important to Safety. The lifting equipment is designed in accordance with ANSI N14.6 as applicable. Other important to safety support equipment is listed in Table 8.1.4.

Auxiliary operational equipment (e.g., trailers, skids, portable cranes, transporters, or air pads) are Not Important to Safety, as the HI-STAR 100 is designed to withstand the failure of any of these components provided the requirements of this FSAR are met.

2.3.3.2 Instrumentation

As a consequence of the passive nature of the HI-STAR 100 System, instrumentation which is Important to Safety is not necessary. No instrumentation is required for HI-STAR 100 storage operations.

2.3.4 Nuclear Criticality Safety

The criticality safety criterion stipulates that the effective neutron multiplication factor, k_{eff}, including statistical uncertainties and biases, is less than 0.95 for all postulated arrangements of fuel within the cask under all credible conditions.

2.3.4.1 Control Methods for Prevention of Criticality

The control methods and design features used to prevent criticality for all MPC configurations are the following:

a. Incorporation of permanent neutron absorbing material in the MPC fuel basket walls.

b. Favorable geometry provided by the MPC fuel basket

Additional control methods used to prevent criticality for the MPC-24 and MPC-32 are the following:

a. Loading of PWR fuel assemblies must be performed in water with a minimum boron content as specified in Table 2.1.18 and 2.1.19, as applicable.

Administrative controls will be used to ensure that fuel placed in the HI-STAR 100 System meets the requirements of the Certificate of Compliance. All appropriate criticality analyses are presented in Chapter 6.

2.3.4.2 Error Contingency Criteria

Provision for error contingency is built into the criticality analyses performed in Chapter 6. Because biases and uncertainties are explicitly evaluated in the analysis, it is not necessary to introduce additional contingency for error.

2.3.4.3 Verification Analyses

In Chapter 6, critical experiments are selected which reflect the design configurations. These critical experiments are evaluated using the same calculation methods, and a suitable bias is incorporated in the reactivity calculation.

2.3.5 Radiological Protection

2.3.5.1 Access Control

As required by 10CFR72, uncontrolled access to the ISFSI will be prevented through physical means. A peripheral fence with an appropriate locking and monitoring system is a standard approach to limit access. The details of the access control systems and procedures, including division of the site into radiation protection areas, *will be developed by the licensee* (user) of the ISFSI utilizing the HI-STAR 100 System.

2.3.5.2 Shielding

The shielding design is governed by 10CFR72.104 and 10CFR72.106 which provide radiation dose limits for any real individual located at or beyond the nearest boundary of the controlled area. The individual must not receive an annual dose equivalent greater than the values stated in Table 2.3.1 for normal and off-normal conditions. Further, an individual located at the site boundary must not receive a dose to the whole body or any organ from any design basis accident greater than the values listed in Table 2.3.1.

The objective of shielding is to assure that radiation dose rates at key locations are as low as practical in order to maintain occupational doses to operating personnel As Low As Reasonably Achievable (ALARA) and to meet the requirements of 10 CFR 72.104 and 10 CFR 72.106 for dose at the controlled area boundary. Three locations are of particular interest in the storage mode:

- immediate vicinity of the cask
- restricted area boundary
- controlled area (site) boundary

Dose rates in the immediate vicinity of the cask are important in consideration of occupational exposure. A design objective for the radial neutron shield mid-height surface dose rate has been established as 130 mrem/h (1.30 mSv/h) . Dose rates above and below the neutron shield may be further reduced by temporary shielding as detailed in Chapter 10. Areas above and below the neutron shield in the radial direction are limited to 375 mrem/h (3.75 mSv/h). Chapter 5 of this FSAR presents the analyses and evaluations to establish HI-STAR 100 compliance with these design objectives.

Because of the passive nature of the HI-STAR 100, human activity related to the system is infrequent and of short duration. Personnel exposures due to operational and maintenance activities are discussed in Chapter 10. Chapter 10 also provides information concerning temporary shielding which may be utilized to reduce the personnel dose during loading, unloading, and handling operations. The estimated occupational doses for personnel must comply with the requirements of 10CFR20.

Dose rates at the restricted area and site boundaries shall be in accordance with applicable regulations. Licensees are required to demonstrate compliance with 10CFR72.104 and 10CFR72.106 for the actual fuel being stored, ISFSI storage array, and controlled area boundary.

The analyses presented in Chapters 5, 10 and 11 demonstrate that the HI-STAR 100 System meets the above radiation dose limits and design objectives.

2.3.5.3 Radiological Alarm System

There are no credible events that could result in release of radioactive materials or increases in direct radiation above the requirements of 10 CFR 72.106.

2.3.6 Fire and Explosion Protection

There are no combustible or explosive materials associated with the HI-STAR 100 System. No such materials would be stored within an ISFSI. However, for conservatism we have analyzed the fire accident as a bounding condition for HI-STAR 100 System. An evaluation of the HI-STAR 100 System in a fire accident is discussed in Chapter 11.

Small overpressures are the result of accidents involving explosive materials which are stored or transported near the site. Explosion is an accident loading condition considered in Chapter 11.

Table 2.3.1

RADIOLOGICAL SITE BOUNDARY REQUIREMENTS

BOUNDARY OF CONTROLLED AREA (m) (minimum)	100
NORMAL AND OFF-NORMAL CONDITIONS: Whole Body - mrem/yr (mSv/yr) Thyroid - mrem/yr (mSv/yr) Any Other Critical Organ - mrem/yr (mSv/yr)	 25 (0.25) 75 (0.75) 25 (0.25)
DESIGN BASIS ACCIDENT: TEDE – rem (Sv) DDE+CDE to any individual organ or tissue (other than lens of the eye) – rem (Sv) Lens dose equivalent – rem (Sv) Shallow dose equivalent to skin or any extremity - rem (Sv)	 5 (0.05) 50 (0.50) 15 (0.15) 50 (0.50)

2.4 DECOMMISSIONING CONSIDERATIONS

Efficient decommissioning of the ISFSI is a paramount objective of the HI-STAR 100 System. The HI-STAR 100 System is ideally configured to facilitate rapid, safe, and economical decommissioning of the storage site.

The MPC which holds the SNF assemblies is engineered to be suitable as a waste package for permanent internment in a deep Mined Geological Disposal System (MGDS). The materials of construction permitted for the MPC are known to be highly resistant to severe environmental conditions. No carbon steel, paint, or coatings are used or permitted in the MPC in areas where they could be exposed to spent fuel pool water or the ambient environment. Therefore, the SNF assemblies stored in the MPC should not need to be removed. However, to ensure a practical, feasible method to defuel the HI-STAR 100 MPC, the top of the MPC is equipped with sufficient gamma shielding and markings locating the drain and vent locations to enable semiautomatic (or remotely actuated) boring of the MPC lid to provide access to the MPC vent and drain. The circumferential welds of the MPC lid and closure ring can be removed by semiautomatic or remotely actuated means, providing access to the SNF.

Likewise, the overpack consists of alloy materials rendering it suitable for permanent burial. Alternatively, the MPC can be removed from the overpack, and the latter reused for storage or transportation of other MPCs.

In either case, the overpack would be expected to have only minimal interior or exterior radioactive surface contamination. Any neutron activation of the metal cask walls and neutron shielding is expected to be extremely small, and the assembly would qualify as Class A waste in a stable form based on definitions and requirements in 10CFR61.55. As such, the material would be suitable for burial in a near-surface disposal site as Low Specific Activity (LSA) material.

If the HI-STAR 100 MPC needs to be opened and separated from the SNF before the fuel is placed into the MGDS, the MPC interior metal surfaces will be decontaminated using existing mechanical or chemical methods. This will be facilitated by the MPC fuel basket and interior structures' smooth metal surfaces designed to minimize crud traps. After the surface contamination is removed, the MPC radioactivity will be diminished significantly, allowing near-surface burial or secondary applications at the licensee's facility.

It is also likely that both the overpack and MPC, or extensive portions of both, can be further decontaminated to allow recycle or reuse options. After decontamination, the only radiological hazard the HI-STAR 100 System may pose is slight activation of the HI-STAR 100 materials caused by irradiation over a 50-year storage period. Table 2.4.1 provides the conservatively determined quantities of the major nuclides after 50 years of irradiation.

The calculation of the material activation is based on the following:

- Beyond design basis fuel assemblies (B&W 15x15, 3.7% enrichment, 47,500 MWD/MTU, and eight-year cooling time) stored for 50 years.
- Cask material quantities based on the design drawings.
- A constant flux equal to the initial loading condition is conservatively assumed.
- Material activation is based on MCNP-4A calculations.

As can be seen by the material activation results presented in Table 2.4.1, the HI-STAR 100 System total activation is very low.

Due to the design of the HI-STAR 100 System, no residual contamination is expected to be left behind on the concrete ISFSI pad. The base pad, fence, and peripheral utility structures will require no decontamination or special handling after the last cask is removed.

In any case, the HI-STAR 100 System would not impose any additional decommissioning requirements on the licensee of the ISFSI facility per 10CFR72.30, since the HI-STAR 100 System could eventually be shipped from the site as a complete assembly.

Table 2.4.1

HI-STAR 100 SYSTEM ACTIVATION

HI-STAR 100 Component	**Nuclide**	**Curies After 50-Year Storage**
MPC	^{54}Mn	1.79E-3
	^{55}Fe	2.88E-3
	^{59}Ni	2.36E-3
	^{60}Co	2.53E-3
	^{63}Ni	8.04E-3
	Total	1.76E-2
Overpack	^{54}Mn	2.24E-3
	^{55}Fe	11.11E-3
	^{59}Ni	N/A
	^{60}Co	N/A
	^{63}Ni	N/A
	Total	1.34E-2
Total HI-STAR 100 System		3.10E-2

2.5 REGULATORY COMPLIANCE

Chapter 2 provides the principal design criteria related to structures, systems, and components important to safety. These criteria include specifications regarding the fuel, as well as, external conditions that may exist in the operating environment during normal and off-normal operations, accident conditions, and natural phenomena events. The chapter has been written to provide sufficient information to allow verification of compliance with 10CFR72, NUREG-1536, and Regulatory Guide 3.61. A more detailed evaluation of the design criteria and an assessment of compliance with those criteria is provided in Chapters 3 through 13.

2.6 REFERENCES

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[2.0.3] American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code", 1995 with Addenda through 1997.

[2.0.4] ANSI N14.6-1993, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More", June 1993.

[2.0.5] ISG- 11, "Cladding Considerations for the Transport and Storage of Spent Fuel," USNRC, Washington, DC, Revision 3, November 17, 2003.

[2.0.6] USNRC Memorandum from Christopher L. Brown to M. Wayne Hodges, "Scoping Calculations for Cladding Hoop Stresses in Low Burnup Fuel," dated January 29, 2004.

[2.0.7] deleted

[2.0.8] *ISG-18, "The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage", USNRC, Revision 1.*

[2.1.1] ORNL/TM-10902, "Physical Characteristics of GE BWR Fuel Assemblies", by R.S. Moore and K.J. Notz, Martin Marietta (1989).

[2.1.2] U.S. DOE SRC/CNEAF/96-01, Spent Nuclear Fuel Discharges from U.S. Reactors 1994, Feb. 1996.

[2.1.3] S.E. Turner, " Uncertainty Analysis - Axial Burnup Distribution Effects," presented in "Proceedings of a Workshop on the Use of Burnup Credit in Spent Fuel Transport Casks", SAND-89-0018, Sandia National Laboratory, Oct., 1989.

[2.1.4] Commonwealth Edison Company, Letter No. NFS-BND-95-083, Chicago, Illinois.

[2.1.5] NUREG-1536, SRP for Dry Cask Storage Systems, USNRC, Washington, DC, January 1997.

[2.1.6] DOE Multi-Purpose Canister Subsystem Design Procurement Specification.

[2.2.1] Crane Manufacturer's Association of America (CMAA), Specification #70, 1988, Section 3.3.

[2.2.2] Cunningham, et als., "Evaluation of Expected Behavior of LWR Stainless-Clad Fuel in Long-Term Dry Storage", EPRI TR-106440, April 1996.

[2.2.3] ASCE 7-88 (formerly ANSI A58.1), "Minimum Design Loads for Buildings and Other Structures", American Society of Civil Engineers, New York, NY, 1990.

[2.2.4] "Debris Collection System for Boiling Water Reactor Consolidation Equipment", EPRI Project 3100-02 and ESEERCO Project EP91-29, October 1995.

[2.2.5] Design Basis Tornado for Nuclear Power Plants, Regulatory Guide 1.76, U.S. Nuclear Regulatory Commission, April 1974.

[2.2.6] ANSI/ANS 57.9-1992, "Design Criteria for an Independent Spent Fuel Storage Installation (dry type)", American Nuclear Society, LaGrange Park, Illinois.

[2.2.7] NUREG-0800, SRP 3.5.1.4, USNRC, Washington, DC.

[2.2.8] 10CFR 100, "Reactor Site Criteria", Title 10 of the Code of Federal Regulations.

[2.2.9] Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants", U.S. Nuclear Regulatory Commission, Washington, D.C., August 1997.

[2.2.10] Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants", U.S. Nuclear Regulatory Commission, Washington, D.C.

[2.2.11] NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety", U.S. Nuclear Regulatory Commission, Washington, D.C., February 1996.

[2.2.12] NUREG/CR-6007, "Stress Analysis of Closure Bolts for Shipping Casks", U.S. Nuclear Regulatory Commission, Washington, D.C., January 1993.

[2.2.13] NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", U.S. Nuclear Regulatory Commission, Washington, D.C., July 1980.

[2.2.14] D. Peckner and I.M. Bernstein, "Handbook of Stainless Steels," McGraw-Hill Book Company, 1977.

[2.2.15] "Nuclear Systems Materials Handbook," Oak Ridge National Laboratory, TID 26666, Volume 1.

CHAPTER 3: STRUCTURAL EVALUATION

3.0 OVERVIEW

In this chapter, the structural components of the HI-STAR 100 System that are important to safety (ITS) are identified and all structural analyses to demonstrate their compliance with the provisions of 10CFR72 are presented. The objective of the structural analyses is to ensure that the integrity of the HI-STAR 100 System is maintained under all credible loads for normal and off-normal conditions, and design basis accident/natural phenomena. The results in this chapter support the conclusion that the confinement, criticality control, radiation shielding, and retrievability criteria set forth by 10CFR72.236(l), 10CFR72.124(a), 10CFR72.104, 10CFR72.106, and 10CFR72.122(l) are met. In particular, the design basis information contained in the previous two chapters and in this chapter provides sufficient data to permit the necessary structural evaluations to demonstrate compliance with the requirements of 10CFR72.24. To facilitate regulatory review, the assumptions and conservatism inherent in the analyses are identified along with a complete description of the analytical methods, models, and acceptance criteria. A summary of other considerations germane to satisfactory structural performance, such as corrosion and material fracture toughness, is also provided.

Detailed numerical computations supporting the conclusions in the main body of this chapter are presented in a series of appendices and calculation packages. Where appropriate, the subsections make reference to results in the appendices and calculation packages. Section 3.6.3 contains the complete list of appendices that support this chapter.

The organization of technical information in this chapter follows the format and content guidelines of USNRC Regulatory Guide 3.61 (February 1989). This revision of this FSAR ensures that the responses to the review requirements listed in NUREG-1536 [2.1.5] are complete and comprehensive. It is noted that the areas of NRC staff technical inquiries, with respect to structural evaluation in NUREG-1536, span a wide array of technical topics within and beyond the material in this chapter. To facilitate staff review to ascertain compliance with the stipulations of NUREG-1536, Table 3.0.1 "Matrix of NUREG-1536 Compliance - Structural Evaluation", is included in this chapter. A comprehensive cross-reference of the topical areas set forth in NUREG-1536 and the location of the required compliance information is contained in Table 3.0.1.

Sections 3.1 through 3.6 provide technical information in the formatting sequence set forth in Regulatory Guide 3.61. Section 3.7 describes in detail HI-STAR 100 System's compliance to NUREG-1536 Structural Evaluation Requirements.

Much of the new information in this chapter is directly extracted from previously NRC approved Holtec dockets; this information is shown in *italics*. In Chapter 3, this information was extracted from References [3.0.1] and [3.0.2]. All changes in this revision are marked with revision bars.

Throughout this chapter SI units[1] are provided in parentheses and rounded to the nearest significant figure consistent with the reported value expressed in US units.

[1] Wherever multiple units are shown, the US units are the governing value and the SI units are shown for information only.

Table 3.0.1 MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION †

PARAGRAPH IN NUREG-1536	NUREG-1536 COMPLIANCE ITEM	LOCATION IN FSAR CHAPTER 3	LOCATION OUTSIDE OF FSAR CHAPTER 3
IV.1.a	ASME B&PV Compliance		
	NB	3.1.1	Tables 2.2.6,2.2.7
	NG	3.1.1	Tables 2.2.6,2.2.7
V.	Identification of SSC that are ITS		Table 2.2.6
“	Applicable Codes/Standards	3.6.1	Table 2.2.6
“	Loads	3.1.2.1.1	Table 2.2.13
“	Load Combinations	3.1.2.1.2; Table 3.1.1, Tables 3.1.3-3.1.5	Table 2.2.14
“	Summary of Safety Factors	3.4.3; 3.4.4.3.1-2; 3.4.6-3.4.8; Tables 3.4.3-3.4.19	
“	Design/Analysis Procedures	Chapter 3	[3.4.13]; [3.4.14]
“	Structural Acceptance Criteria		Tables 2.2.10-2.2.12
“	Material/QC/Fabrication	Table 3.4.2	Chap. 9; Chap. 13
“	Testing/In-Service Surveillance		Chap. 9; Chap. 12

Table 3.0.1 MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION † (Continued)

PARAGRAPH IN NUREG-1536	NUREG-1536 COMPLIANCE ITEM	LOCATION IN FSAR CHAPTER 3	LOCATION OUTSIDE OF FSAR CHAPTER 3
"	Conditions for Use		Table 1.2.6; Chaps. 8,9,12
V.1.a	Description of SSC	3.1.1	1.2; Table 2.2.6
V.1.b.i.(2)	Identification of Codes & Standards		Tables 2.2.6, 2.2.7
V.1.b.ii	Drawings/Figures		1.5
"	Identification of Confinement Boundary		1.5; 2.3.2; 7.1; Table 7.1.1
"	Boundary Weld Specifications	3.3.1.4	1.5; Table 7.1.2
"	Boundary Bolt Torque	NA	
"	Weights and C.G. Location	Tables 3.2.1-3.2.4	
"	Chemical/Galvanic Reactions	3.4.1; Table 3.4.2	
V.1.c	Material Properties	3.3; Tables 3.3.1-3.3.5	1.A; Figures 1.A.1-1.A-5; 1.C
"	Allowable Strengths	3.1.2.2; Tables 3.1.6-3.1.17	Tables 2.2.10-2.2.12
"	Suitability of Materials	3.3; Table 3.4.2	1.A; 1.B
"	Corrosion	3.4.1; Table 3.4.2	
"	Material Examination before Fabrication		9.1.1

Table 3.0.1 MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION † (Continued)

PARAGRAPH IN NUREG-1536	NUREG-1536 COMPLIANCE ITEM	LOCATION IN FSAR CHAPTER 3	LOCATION OUTSIDE OF FSAR CHAPTER 3
"	Material Testing and Analysis		9.1; Table 9.1.1; Table 9.1.2
"	Material Traceability		9.1.1
"	Material Long Term Performance	3.4.10; 3.4.11	9.2
"	Materials Appropriate to Load Conditions		Chap. 1
"	Restrictions on Use		Chap. 12
"	Temperature Limits	Table 3.1.17	Table 2.2.3
"	Creep/Slump	NA	
"	Brittle Fracture Considerations	3.1.2.3; Table 3.1.18-19	
"	Low Temperature Handling	NA	NA
V.1.d.i.(1)	Normal Load Conditions		2.2.1; Tables 2.2.13,2.2.14
"	Fatigue	3.1.2.4	
"	Internal Pressures/Temperatures for Hot and Cold Conditions	3.4.4.1	Tables 2.2.1,2.2.3
"	Required Evaluations		

Table 3.0.1 MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION † (Continued)

PARAGRAPH IN NUREG-1536	NUREG-1536 COMPLIANCE ITEM	LOCATION IN FSAR CHAPTER 3	LOCATION OUTSIDE OF FSAR CHAPTER 3
"	Weight+Pressure	3.4.4.3.1.2; Table 3.4.7	
"	Weight+Pressure+Temp.	3.4.4.3.1.2; Table 3.4.8	
"	Free Thermal Expansion	3.4.4.2	Tables 4.4.16, 4.4.22; Table 11.2.5
V.1.d.i.(2)	Off-Normal Conditions		2.2.2; Tables 2.2.13, 2.2.14; 11.1
V.1.d.i.(3)	Accident Level Events and Conditions	Table 3.1.2	2.2.3; Tables 2.2.13, 2.2.14; 11.2
V.1.d.i.(3).(a)	Storage Cask Vertical Drop	3.1.2.1.1.2; 3.4.9; 3.A	
"	Storage Cask Tipover	3.1.2.1.1.1; 3.4.9; 3.A	2.2.3.2
"	Storage Horizontal Drop	3.1.2.1.1.2; 3.A	
V.1.d.i.(3).(b)	Explosive Overpressure	3.1.2.1.1.4	2.2.3.10
V.1.d.i.(3).(c)	Fire		
"	Structural Evaluations	3.4.4.2.2	2.2.3.3
"	Material Properties	3.4.4.2.2	
"	Material Suitability	3.3.1.1; 3.4.4.2.2	Table 2.2.3
V.1.d.i.(3).(d)	Flood		

Table 3.0.1 MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION † (Continued)

PARAGRAPH IN NUREG-1536	NUREG-1536 COMPLIANCE ITEM	LOCATION IN FSAR CHAPTER 3	LOCATION OUTSIDE OF FSAR CHAPTER 3
"	Identification	3.1.2.1.1.3; 3.4.6	2.2.3.6
"	Cask Tipover	3.4.6	
"	Cask Sliding	3.4.6	
"	Hydrostatic Loading	3.1.2.1.1.3; 3.4.6	Supplement 23 of [3.4.13]
"	Consequences	3.4.6	11.2
V.1.d.i.(3).(e)	Tornado Winds		
"	Specification	3.1.2.1.1.5	2.2.3.5; Table 2.2.4
"	Drag Coefficients	3.4.8	Supplement 19 of [3.4.13]
"	Load Combination	3.4.8	Supplement 19 of [3.4.13]
"	Overturning	3.4.8	Supplement 19 of [3.4.13]
"	Overturning –Transfer Cask	NA	
V.1.d.i.(3).(f)	Tornado Missiles		
"	Missile Parameters	3.1.2.1.1.5	Table 2.2.5
"	Tipover	3.4.8	Supplement 19 of [3.4.13]
"	Damage		Supplements 22 and 23 of [3.4.13]

Table 3.0.1 MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION † (Continued)

PARAGRAPH IN NUREG-1536	NUREG-1536 COMPLIANCE ITEM	LOCATION IN FSAR CHAPTER 3	LOCATION OUTSIDE OF FSAR CHAPTER 3
"	Consequences	3.4.8	11.2
V.1.d.i.(3).(g)	Earthquakes		
"	Definition of DBE	3.1.2.1.1.6; 3.4.7	2.2.3.7; Table 2.2.8
"	Sliding	3.4.7.1	
"	Overturning	3.4.7.1	
"	Structural Evaluations	3.4.7.2	
V.1.d.i.(4).(a)	Lifting Analyses		
"	Trunnions		
"	Requirements	3.1.1; 3.4.3.1	
"	Analyses	3.4.3.1	Supplement 20 of [3.4.13]
"	Other Lift Analyses	3.4.3.2-3.4.3.4	Supplements 20 and 26 of [3.4.13]; supplements 14, 15, 52, and 48 of [3.4.14]
V.1.d.i.(4).(b)	Fuel Basket		
"	Requirements	3.1.2.1.2; Table 3.1.3	

Table 3.0.1 MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION † (Continued)

PARAGRAPH IN NUREG-1536	NUREG-1536 COMPLIANCE ITEM	LOCATION IN FSAR CHAPTER 3	LOCATION OUTSIDE OF FSAR CHAPTER 3
"	Specific Analyses	3.4.4.2; 3.4.4.3.1; 3.4.4.4.1; Table 3.4.3; Table 3.4.9; Table 3.4.11	Supplements 51, 18, and 21 of [3.4.14]
"	Dynamic Amplifiers		Supplement 25 of [3.4.13]
"	Stability	3.4.4.3.1.3; Figures 3.4.27-32	
V.1.d.i.(4).(c)	Confinement Closure Lid Bolts		
"	Pre-Torque	NA	
"	Analyses	NA	
"	Engagement Length	NA	
"	Miscellaneous Bolting		
"	Pre-Torque		Table 8.1.3; Supplement 21 of [3.4.13]; Supplement 52 of [3.4.14]
"	Analyses	3.4.4.3.2.3; Tables 3.4.17, 3.4.18	Supplements 21, 27, 28 and 29 of [3.4.13]; Supplement 52 of [3.4.14]
"	Engagement Length		Supplement 52 of [3.4.14]

Table 3.0.1 MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION † (Continued)

PARAGRAPH IN NUREG-1536	NUREG-1536 COMPLIANCE ITEM	LOCATION IN FSAR CHAPTER 3	LOCATION OUTSIDE OF FSAR CHAPTER 3
V.1.d.i.(4)	Confinement		
"	Requirements	3.1.2.1.2; Table 3.1.4	Chap. 7
"	Specific Analyses	3.4.4.2; 3.4.4.3.1; 3.4.4.4.1; Tables 3.4.4, 3.4.7-3.4.9, 3.4.11-3.4.15	Chap. 7; Supplements 14, 15 and 28 of [3.4.14]; Supplements 23 and 26 of [3.4.13]
"	Dynamic Amplifiers		Supplement 25 of [3.4.13]
"	Stability	3.4.4.3.1.7	Supplement 23 of [3.4.13]

Table 3.0.1 MATRIX OF NUREG-1536 COMPLIANCE ITEMS – STRUCTURAL EVALUATION † (Continued)

PARAGRAPH IN NUREG-1536	NUREG-1536 COMPLIANCE ITEM	LOCATION IN FSAR CHAPTER 3	LOCATION OUTSIDE OF FSAR CHAPTER 3
"	Overpack		
"	Requirements	3.1.2.1.2; Tables 3.1.1, 3.1.5	
"	Specific Analyses	3.4.4.3.2; 3.4.4.4.2; 3.L; Tables 3.4.5, 3.4.6, 3.4.10, 3.4.16	Supplements 26 and 30 of [3.4.13]
"	Dynamic Amplifiers		Supplement 25 of [3.4.13]
"	Stability	3.4.4.3.2.5	Supplement 23 of [3.4.13]

† Legend for Table 3.0.1

Per the nomenclature defined in Chapter 1, the first digit refers to the chapter number, the second digit is the section number within the chapter; an alphabetic character in the second place means it is an appendix to the chapter.

NA Not Applicable for this item

3.1 STRUCTURAL DESIGN

3.1.1 Discussion

The HI-STAR 100 System consists of two principal components: the Multi-Purpose Canister (MPC), and the overpack. The MPC is a hermetically sealed, welded structure of cylindrical profile with flat ends and a honeycomb fuel basket. A complete description is provided in Section 1.2.1.1 wherein the anatomy of the MPC and its fabrication details are presented with the aid of figures. A detailed discussion of the HI-STAR 100 overpack geometry is presented in Section 1.2. Drawings for the HI-STAR 100 System are provided in Section 1.5. In this section, the discussion is confined to characterizing and establishing the structural features of the MPC and the storage overpack.

The design of the MPC seeks to attain three objectives which are central to its functional adequacy, namely;

- Ability to Dissipate Heat: The thermal energy produced by the stored spent fuel must be transported to the outside surface of the MPC such that the prescribed temperature limits for the fuel cladding and for the fuel basket metal walls are not exceeded.

- Ability to Withstand Large Impact Loads: The MPC, with its payload of nuclear fuel, must be sufficiently robust to withstand large impact loads associated with the postulated handling accident events. Furthermore, the strength of the MPC must be sufficiently isotropic to meet structural requirements under a variety of handling and tip-over accidents.

- Restraint of Free End Expansion: The membrane and bending stresses produced by restraint of free-end expansion of the fuel basket are conservatively categorized as primary stresses. In view of the concentration of heat generation in the fuel basket, it is necessary to ensure that structural constraints to its external expansion do not exist.

Where the first two criteria call for extensive inter-cell connections, the last criterion requires the opposite. The design of the MPC seeks to realize all of the above three criteria in an optimal manner.

As the description presented in Chapter 1 indicates, the MPC enclosure vessel is the confinement vessel designed to meet ASME Code, Section III, Subsection NB stress limits. The enveloping canister shell, the baseplate, and the lid system form a complete confinement boundary for the stored fuel which is referred to as the "enclosure vessel". Within this cylindrical shell confinement vessel is an integrally welded assemblage of cells comprised of square cross sectional openings for fuel storage, referred to herein as the "fuel basket". The fuel basket is analyzed under the provisions of Subsection NG of Section III of the ASME Code. There are three different multi-purpose canisters which are exactly alike in their external dimensions. The essential difference between the MPCs lies in the fuel baskets. Each fuel storage MPC is designed to house fuel assemblies with different characteristics. Although all fuel baskets are configured to maximize structural ruggedness through extensive inter-cell connectivity, they are sufficiently dissimilar in structural details to warrant separate evaluations. Therefore, analyses for each of the three MPC types are presented, as appropriate, throughout this chapter.

Components of the HI-STAR 100 System that are important to safety and their applicable design codes are defined in Chapter 2.

The structural function of the MPC in the storage mode is:

1. To maintain position of the fuel in a subcritical configuration, and

2. To provide a radiological confinement boundary.

The structural function of the overpack in the storage mode is:

1. To serve as a missile barrier for the MPC

2. To ensure stability of the HI-STAR 100 System, and

3. To provide a structurally robust support for the radiation shielding, and

4. To provide a helium retention boundary

Some structural features of the MPCs which allow the system to perform these functions are summarized below:

- There are no gasketed ports or openings in the MPC. The MPC does not rely on any sealing arrangement except welding. The absence of any gasketed or flanged joints precludes joint leaks. The confinement boundary contains no valves or other pressure relief devices.

- The closure system for the MPCs consists of two components, namely, the MPC lid and closure ring. The MPC lid can be either a single thick lid or two dual lids welded around their common periphery. The MPC closure system is shown in the drawings in Section 1.5. The MPC lid-to-MPC shell weld is a J-groove weld which is subject to root and final pass liquid penetrant examinations and finally, a volumetric examination to ensure the absence of unacceptable flaws and indications. The MPC lid is equipped with vent and drain ports which are utilized for evacuating moisture and air from the MPC following fuel loading, and subsequent backfilling with an inert gas (helium) in a specified quantity. The vent and drain ports are covered by a cover plate and welded before the closure ring is installed. The closure ring is a circular annular plate edge-welded to the MPC shell. The two closure members are interconnected by welding around the inner diameter of the ring. Lift points for the MPC are provided in the MPC lid.

- The MPC fuel baskets consist of an array of interconnecting plates (Figure 3.1.1). The number of storage cells formed by this interconnection process varies depending on the type of fuel being stored. Basket designs containing 24 (PWR), 32 (PWR), and 68 (BWR) cell configurations have been designed and are explained in detail in Section 1.2. All baskets are designed to fit into the same MPC shell. Welding of the basket plates along their edges essentially renders the fuel basket into a multiflange beam. Figure 3.1.1 provides an isometric illustration of a fuel basket for the MPC-68 design.

- The MPC basket is separated from the longitudinal supports installed in the enclosure vessel shell by a small gap. The gap size decreases as a result of thermal expansion (depending on the magnitude of internal heat generation from the stored spent fuel). The provision of a small gap between the basket and the basket support structure is consistent with the natural thermal characteristics of the MPC. The planar temperature distribution across the basket, as shown in Section 4.4, approximates a shallow parabolic profile. This profile will create high thermal stresses unless structural constraints at the interface between the basket and the basket support structure are removed.

The MPCs will be loaded with fuel with widely varying heat generation rates. The basket/basket support structure gap tends to be reduced due to thermal expansion from decay heat generation. Gaps between the fuel basket and the basket support structure are specified to be sufficiently large such that a gap will exist around the periphery under any normal or off-normal operating or accident conditions (such as the postulated fire event).

A small number of flexible thermal conduction elements (thin aluminum tubes) are interposed between the basket and the MPC shell. The elements are designed to be resilient. They do not provide structural support for the basket, and thus their resistance to thermal growth is negligible. It is quite evident from the geometry of the MPC that a critical loading event pertains to the drop condition, when the MPC is postulated to undergo a handling side drop (the longitudinal axis of the MPC is horizontal) or tip-over. Under the side drop or tip-over condition the flat panels of the fuel basket are subject to an equivalent pressure loading that simulates the deceleration magnified inertia load from the stored fuel and the MPC's own metal mass.

The MPC fuel basket maintains the spent nuclear fuel in a subcritical arrangement. Its safe operation is assured by maintaining the physical configuration of the storage cell cavities intact in the aftermath of a drop event. This requirement is considered to be satisfied if the MPC fuel basket meets the stress intensity criteria set forth in the ASME Code, Section III, Subsection NG. Therefore, the demonstration that the fuel basket meets Subsection NG limits ensures that there is no impairment of ready retrievability (as required by NUREG-1536), that there is no unacceptable release of radioactive materials, and that there is no unacceptable radiation level.

The MPC confinement boundary contains no valves or other pressure relief devices. The MPC enclosure vessel will be shown to meet the stress intensity criteria of the ASME Code, Section III, Subsection NB for all service conditions.

Structural features of the overpack that allow the system to perform its structural function are summarized below:

- The HI-STAR 100 overpack is a missile barrier, radiation shield, and helium retention boundary in the storage mode. The overpack provides kinematic stability to the system, and acts as a cushion for the MPC in the event of a postulated tip-over accident. The overpack features a thick inner shell welded to a bottom plate which forms the load bearing surface (foundation interface) for the HI-STAR 100 System. A solid metal top flange welded at the top of the inner shell provides the attachment location for lifting trunnions. The top flange is also designed to provide a recessed ledge for the closure plate to protect the bolts from direct shear loadings resulting from an impulsive load at the top edge of the overpack. The helium retention boundary of the HI-STAR 100 overpack is subject to the stress limits of the ASME Code, Section III, Subsection NB.

- The inner shell is reinforced by multilayered intermediate shells. The multi layer approach eliminates the potential for a crack in any one layer, developed by any postulated mechanical loading or material flaw, to travel uninterrupted through the vessel wall. The intermediate shells also buttress the overpack inner shell against buckling. The intermediate shells of the HI-STAR 100 overpack are subject to the stress limits of the ASME Code, Section III, Subsection NF, Class 3.

- To facilitate handling of the loaded system, the HI-STAR 100 overpack is equipped with lifting trunnions and pocket trunnions (optional). The pocket trunnions (optional) are located on the overpack intermediate shells just above the bottom plate. The centerline through the pocket trunnion recess is offset from a vertical plane containing the overpack's center of gravity to ensure a stable rotation direction during upending and down ending operations. Lifting trunnions are conservatively designed to meet the design safety factor requirements of NUREG-0612 [3.1.1] and ANSI N14.6-1993 [3.1.2] for single failure proof lifting equipment.

Table 2.2.6 provides a listing of the applicable design codes for all structures, systems, and components which are designated as “Important to Safety” (ITS). Since no structural credit is required for the weld between the adjustable basket support pieces (i.e., shims and basket support flat plates), the adjustable basket supports are classified as NITS.

3.1.2 Design Criteria

This subsection provides information requested in Subsection 3.1.2 of Regulatory Guide 3.61. Principal design criteria for normal, off-normal, and accident/environmental events are discussed in Section 2.2 in Chapter 2. In this section, the loads, load combinations, and allowable stresses used in the structural evaluation of the HI-STAR 100 System are presented.

Consistent with the provisions of NUREG-1536, the central objective of the structural analysis presented in this chapter is to ensure that the HI-STAR 100 System possesses sufficient structural capability to withstand normal and off-normal loads and the worst case loads under natural phenomenon events. Withstanding such loadings enables the HI-STAR 100 System to successfully preclude the following negative consequences:

- risk of criticality
- release of radioactive materials
- unacceptable radiation levels
- impairment of ready retrievability of the SNF

The design objectives for the HI-STAR 100 are particularized for individual components as follows:

- The objective of the structural analysis of the MPC is to demonstrate that:

    1. Confinement of radioactive material is maintained under normal, off-normal, accident conditions, and natural phenomenon events.

    2. The MPC basket does not deform under credible loading conditions such that the sub-criticality or retrievability of the SNF is jeopardized.

- The objective of the structural analysis of the storage overpack is to demonstrate that:

    1. Tornado-generated missiles do not compromise integrity of the MPC confinement boundary.

    2. The integrity of the helium retention boundary is not compromised.

    3. The radiation shielding remains properly positioned in the case of a natural phenomenon or accident event.

The aforementioned objectives are deemed to be satisfied for the MPC, and the overpack, if stresses (or stress intensities, as applicable) calculated by the appropriate structural analyses are less than the allowables defined in Subsection 3.1.2.2.

Stresses arise in the components of the HI-STAR 100 System due to various loads which originate under normal, off-normal, or accident conditions. These individual loads are combined to form load combinations. Stresses and stress intensities resulting from the load combinations are compared to allowable stresses and stress intensities. The following subsections present loads, load combinations, and allowable strengths for use in the structural analyses of the MPC and the HI-STAR 100 overpack.

3.1.2.1 Loads and Load Combinations

The individual loads are defined in Section 2.2 of this report (Table 2.2.13). Load combinations are developed by appropriately combining the individual loads (Table 2.2.14). Load combinations are applied to the mathematical models of the MPCs, and the overpack. Results of the analyses carried out under bounding load combinations are compared to allowable stresses or stress intensities, as applicable.

3.1.2.1.1 Individual Load Cases

The individual load cases which address each design criterion applicable to the structural design of the HI-STAR 100 System are catalogued in Table 2.2.13. Each load is given a symbol for subsequent use in the load combination listed in Table 2.2.14.

Accident condition and natural phenomena-induced events, collectively referred to as the "Level D" condition in Section III of the ASME Boiler & Pressure Vessel Codes, *in general* do not have a universally prescribed limit. For example, the impact load from a tornado borne missile, or the overturning load under flood or tsunami, cannot be prescribed as design basis values with absolute certainty that all ISFSI sites will be covered. Therefore, as applicable, allowable magnitudes of such loadings are postulated for the HI-STAR system. The allowable values are drawn from ANSI documents (such as for tornado and wind) or from an intrinsic limitation in the system (such as the permissible "drop height" under a postulated handling accident). In the following, the essential characteristic of each "Level D" type loading is explained.

3.1.2.1.1.1 Tip-Over

It is required to demonstrate that the HI-STAR 100 System will not tip over as a result of a postulated natural phenomenon event, including tornado wind, a tornado-generated missile, a seismic or a hydrological event (flood). However, to demonstrate the defense-in-depth features of the design, a non-mechanistic tip-over scenario per NUREG-1536 (page 3-11) is analyzed. Table 3.1.2 lists the design basis deceleration limit.

If a non-single failure proof lifting device is employed to carry the cask over the ISFSI pad, determination of maximum carry height must be performed by the ISFSI owner once the ISFSI pad design is formalized.

3.1.2.1.1.2 Handling Accident

The design basis handling accident during transport of a loaded HI-STAR 100 storage overpack results in either a vertical or horizontal drop. Table 3.1.2 lists the design basis deceleration limit. In addition, the handling of all heavy loads that are within Part 72 jurisdiction should be carried out using single-failure-proof equipment and lifting devices that comply with the stress limits of ANSI N14.6 to render an uncontrolled lowering of the payload non-credible.

If a non-single failure proof lifting device is employed to carry the cask over the ISFSI pad, determination of maximum carry height must be performed by the ISFSI owner once the ISFSI pad design is established.

3.1.2.1.1.3 Flood

The postulated flood event results into two discrete scenarios which must be considered; namely,

1. Stability of the HI-STAR 100 System due to flood water velocity, and

2. Structural effects of hydrostatic pressure and water velocity induced lateral pressure.

The maximum design external pressure for the overpack is 300 psi (2 MPa) (Table 2.2.1). The maximum design flood water depth of 656 ft (200 m) (Table 2.2.8) corresponds to an external pressure that is bounded by the design external pressure in Table 2.2.1.

3.1.2.1.1.4 Explosion

Explosive materials are not permitted within the protective boundary of an ISFSI where a loaded HI-STAR 100 System is maintained in normal storage. The accident condition overpack external pressure specified in Table 2.2.1 is also set as the overall external pressure that bounds all credible external explosion events. There are no credible internal explosion events.

3.1.2.1.1.5 Tornado

The three components of a tornado load are:

1. Pressure changes,
2. Wind loads, and
3. Tornado-generated missiles.

Wind speeds and tornado-induced pressure drop are specified in Table 2.2.4. Tornado missiles are listed in Table 2.2.5. Potential consequences of a tornado on the cask system are:

- Instability (tip-over) due to tornado missile impact plus either steady wind or impulse from sudden pressure drop.

- Stress in the overpack induced by the lateral force caused by the steady wind or missile impact.

3.1.2.1.1.6 Earthquake

Subsections 2.2.3.7 and 3.4.7 contain the detailed specification of the seismic inputs applied to the HI-STAR 100 System. The design basis earthquake is assumed to be applied at the top of the ISFSI pad. Potential consequences of a seismic event are sliding/overturning of the loaded overpack, and stresses in the overpack arising from the inertia forces on the system.

3.1.2.1.1.7 Lightning

The HI-STAR 100 overpack contains many thousands of pounds of highly conductive carbon steel with over 400 square feet of external surface area. Such a large surface area and metal mass is adequate to dissipate any lightning which may strike the HI-STAR 100 System. There are no combustible materials on the HI-STAR 100 surface. Therefore, lightning will not impair the structural performance of components of the HI-STAR 100 System that are important to safety.

3.1.2.1.2 Load Combinations

Load combinations are created by summing the effects of all applicable individual loads which can act concurrently. The load combinations are selected for the normal, off-normal, and accident conditions. The loadings appropriate for HI-STAR 100 under the various conditions are presented in Table 2.2.14. These loadings are combined into meaningful combinations for the various HI-STAR 100 System components in Tables 3.1.1, and 3.1.3-3.1.5. Table 3.1.1 lists the load combinations that address overpack stability. Tables 3.1.3 through 3.1.5 list the applicable load combinations for the fuel basket, the enclosure vessel, and the overpack, respectively.

As discussed in Section 2.2.7, the number of discrete load combinations for each situational condition (i.e., normal, off-normal, etc.) is consolidated by defining bounding loads for certain groups of loadings. Thus, the accident condition pressure P_o^* bounds the surface loadings arising from accident and extreme natural phenomenon events, namely tornado wind W', flood F, and explosion E^*.

As noted previously, certain loads, namely earthquake E, flowing water under flood condition F, and tornado missile M, act to destabilize a cask. Additionally, these loads act on the overpack and produce localized stresses at the HI-STAR 100 System to ISFSI interface. Table 3.1.1 provides the load combinations which are relevant to the stability analyses. The site ISFSI DBE zero period acceleration (ZPA) must be bounded by the design basis seismic ZPA defined by the Load Case C of Table 3.1.1 to demonstrate that the margins against tip-over and inter-cask collision during a seismic event are maintained.

As noted at the beginning of this section, there are two principal components to the HI-STAR 100 System: the multi-purpose canister (MPC) and the overpack. The MPC is made up of the fuel basket and the enclosure vessel. A complete account of analyses and results for all load cases for all three constituent parts: (i) the fuel basket, (ii) the enclosure vessel, and (iii) the overpack is provided in Section 3.4, as required by Regulatory Guide 3.61.

In the following, the loadings listed as applicable for each situational condition in Table 2.2.14 are addressed in meaningful load combinations for the fuel basket, enclosure vessel, and the overpack. Each component is considered separately. It is noted that off-normal condition pressure temperatures for structural analyses are conservatively bounded by the specified design pressures and temperatures. Therefore, load combinations for normal and off-normal condition are subsumed into a consolidated set of bounding load combinations.

Fuel Basket

Table 3.1.3 summarizes all loading cases (derived from Table 2.2.14) which are germane to demonstrating compliance of the fuel baskets to Subsection NG when these baskets are housed within the HI-STAR 100 overpack.

Normal Condition

- The fuel basket is not a pressure vessel; therefore, the pressure loadings are not meaningful loads for the basket. Further, the basket is structurally decoupled from the enclosure vessel. The gap between the basket and the enclosure vessel is sized to ensure that no constraint of free-end thermal expansion of the basket occurs. The demonstration of the adequacy of the basket to the enclosure vessel (EV) gap to ensure absence of interference is a physical problem which must be analyzed. Temperature, like pressure, is not a source of loading for the fuel basket. All loadings on the fuel basket, therefore, arise from handling and postulated handling accident conditions.

- Normal handling encompasses both vertical and horizontal orientation. When the cask is being handled in the vertical orientation, the vertical load produces a strictly axial compressive stress. When the cask is being lifted from the horizontal orientation, the amplified dead load may cause flexing of the fuel basket panels.

Off-Normal Conditions

- The off-normal condition handling loads are identical to the normal condition, and therefore, a separate analysis is not required.

Accident Condition

- Three accident condition scenarios must be considered: (i) drop with the storage overpack axis vertical; (ii) drop with the storage overpack axis horizontal; and (iii) storage overpack tip-over.

- The vertical drop scenario induces compression in the longitudinal panel of the fuel basket.

- The horizontal drop and tip-over must consider multiple orientations of the fuel basket as the fuel basket is not radially symmetric. Heretofore, two horizontal drop orientations are considered which are referred to as the 0-degree drop and 45-degree drop, respectively. In the 0-degree drop, the basket drops with its panels oriented parallel and normal to the vertical (see Figure 3.1.2). The 45-degree drop implies that the basket's honeycomb section is rotated meridionally by 45 degrees (Figure 3.1.3).

Enclosure Vessel

Table 3.1.4 summarizes all load cases that are applicable to structural analysis of the enclosure vessel to ensure integrity of the confinement boundary.

Normal Conditions

- The enclosure vessel is a pressure vessel consisting of a cylindrical shell, a thick circular baseplate at the bottom, and a thick circular lid at the top. This pressure vessel must be shown to meet the primary stress intensity limits for ASME Section III Class 1 at the design temperature and primary plus secondary stress intensity limits under the combined action of pressure plus thermal loads.

- The MPC lid system of the enclosure vessel is equipped with tapped holes for lifting operations. A normal handling operation is defined to encompass a vertical lift where the MPC is supported by threaded inserts in the MPC lid. Stress intensities in the MPC lid, must satisfy Level A limits for Class 1 components. The threaded inserts (i.e., the lifting eye bolts) and the internal threads in the tapped holes must meet NUREG-0612 stress limits. Further discussion on design criteria applicable to lifting operations is presented in Subsection 3.4.3 herein, as required by Regulatory Guide 3.61.

Off-Normal Conditions

- The off-normal condition loads are identical to the normal condition, and therefore, a separate analysis is not required.

Accident Conditions

- The design basis deceleration for the MPC in the HI-STAR 100 System is 60g's. The deceleration loading developed in the enclosure vessel during a horizontal drop event must be combined with those due to P_i (internal pressure) acting alone. The accident condition pressure is bounded by P_i^*. During a vertical drop scenario, the axial buckling of the enclosure vessel shell is the item of principal concern. To render the loading combination most adverse, the vertical deceleration load is assumed to act in the absence of P_i, which produces tensile stresses (and thus counteracts the loads which produce buckling).

- The fire event (T^* loading) is considered for ensuring absence of interference between the enclosure vessel and the fuel basket and between the enclosure vessel and the overpack. The metal temperatures of the "NB pressure parts" (defined by the ASME Code, loc. cit.) are required to remain in the range of temperatures permitted by the ASME Code.

Storage Overpack

Table 3.1.5 identifies the load cases to be considered for the overpack. These are in addition to the kinematic criteria listed in Table 3.1.1. Within these load cases and kinematic criteria, the following items must be addressed:

Normal Conditions

- The inner shell, the bottom plate, the top flange, and the closure plate of the overpack constitute a pressure vessel and must be engineered to meet ASME Code requirements for helium pressure retention.

- In the normal handling condition, the most adverse configuration is the vertical lift. The top flange/closure plate region and the bottom plate are most affected by the handling loads acting in concert with design internal or external pressure. The specific stress limits which must be satisfied under normal handling are discussed in depth in Subsection 3.4.3, as required by Regulatory Guide 3.61.

Off-Normal Conditions

- The off-normal condition loads are identical to the normal condition, and therefore, a separate analysis is not required.

Accident Conditions

- Maximum flood water velocity for the overpack with an empty MPC (to minimize system weight and thus maximize the potential for kinematic instability) must be specified to ensure that no sliding or tip-over occurs.

- Tornado missile plus wind on an overpack with an empty MPC must be specified to demonstrate that no cask tip-over occurs.

- Tornado missile penetration analysis must demonstrate that the postulated penetrant missiles cannot reach the MPC stored inside the HI-STAR 100 overpack.

- Under seismic conditions, a fully loaded HI-STAR 100 overpack must not tip over under the maximum ZPA event. The maximum sliding of the overpack must demonstrate that casks will not impact each other.

- Under a non-mechanistic postulated tip-over or a drop accident with a full HI-STAR 100 overpack, the overpack structure must meet faulted (Level D) requirements of the ASME Code.

Finally, it is noted that the structural evaluations performed for the HI-STAR 100 System under normal and off-normal conditions are bounding regardless of whether the overpack is stored vertically or horizontally. This is because of the following considerations.

i) The normal design pressures and temperatures listed in Tables 2.2.1 and 2.2.3 of this FSAR, which are used as input to the structural evaluations, are bounding for both storage orientations (horizontal and vertical) as confirmed by the thermal evaluations in Chapter 4.

ii) As stated above, normal handling encompasses both vertical and horizontal orientations (regardless of the final storage configuration). Accordingly, the MPC fuel basket and enclosure shell are analyzed separately in Subparagraph 3.4.4.3.1 under axial and lateral loads due to normal handling. The evaluations for the HI-STAR lifting trunnion, pocket trunnions, and the various lid lifting points are not affected by the final storage configuration.

iii) As explained in Section 11.1 of this FSAR, the maximum component temperatures due to off-normal environmental conditions, when the HI-STAR 100 overpack is stored horizontally, are bounded by the results presented in Chapter 3 of the HI-STAR 100 SAR for normal conditions of transport. Moreover, the bounding component temperatures reported in Tables 3.4.10, 3.4.17, and 3.4.18 of the HI-STAR 100 SAR are less than the design temperatures listed in Table 2.2.3 of this FSAR for normal conditions of storage. In addition, the off-normal design pressures are the same as the normal design pressures, as shown in Table 2.2.1 and further discussed in Paragraph 11.1.1.3, so that the load combinations for normal and off-normal conditions are subsumed into a bounding set of load combinations.

In summary, the structural evaluations for the normal and off-normal load combinations are bounding for a loaded HI-STAR 100 overpack stored in the vertical or horizontal orientation.

3.1.2.2 Allowables

The important to safety components of the HI-STAR 100 System are listed in Table 2.2.6. Allowable stresses, as appropriate, are tabulated for these components for all service conditions in Tables 3.1.6 through 3.1.16.

In Subsection 2.2.5, the applicable service level from the ASME Code for determination of allowables is listed. Table 2.2.14 provides a tabulation of normal, off-normal, and accident conditions and the service levels defined in the ASME Code, along with the applicable loadings for each service condition.

Allowable stresses and stress intensities are calculated using the data provided in the ASME Code and Tables 2.2.10 through 2.2.12. Tables 3.1.6 through 3.1.16 contain numerical values of the stresses/stress intensities for all MPC and overpack load bearing materials as a function of temperature.

In all tables the terms S, S_m, S_y, and S_u, respectively, denote the design stress, design stress intensity, minimum yield strength, and the ultimate strength. Property values at intermediate temperatures which are not reported in the ASME Code are obtained by linear interpolation. Property values are not extrapolated beyond the limits of the Code in any structural calculation.

Additional terms relevant to the analyses are extracted from the ASME Code (Figure NB-3222-1, for example) as follows:

Symbol	Description	Notes
P_m	Average primary stress across a solid section.	Excludes effects of discontinuities and concentrations. Produced by pressure and mechanical loads.
P_L	Average stress across any solid section.	Considers effects of discontinuities but not concentrations. Produced by pressure and mechanical loads, including inertia earthquake effects.
P_b	Primary bending stress.	Component of primary stress proportional to the distance from the centroid of a solid section. Excludes the effects of discontinuities and concentrations. Produced by pressure and mechanical loads, including inertia earthquake effects.
P_e	Secondary expansion stress.	Stresses which result from the constraint of free-end displacement. Considers effects of discontinuities but not local stress concentration. (Not applicable to vessels.)
Q	Secondary membrane plus bending stress.	Self-equilibrating stress necessary to satisfy continuity of structure. Occurs at structural discontinuities. Can be caused by pressure, mechanical loads, or differential thermal expansion.
F	Peak stress.	Increment added to primary or secondary stress by a concentration (notch), or, certain thermal stresses which may cause fatigue but not distortion. This value is not used in the tables.

It is shown that there is no interference between component parts due to free thermal expansion. Therefore, P_e does not develop within any HI-STAR 100 component.

It is recognized that the planar temperature distribution in the fuel basket and the overpack under the maximum heat load condition is the highest at the cask center and drops monotonically, reaching its lowest value at the outside surface. Strictly speaking, the allowable stresses/stress intensities at any location in the basket, the enclosure vessel, or the overpack should be based on the coincident metal temperature under the specific operating condition. However, in the interest of conservatism, reference temperatures are established for each component which are upper bound on the metal temperature for each situational condition. Table 3.1.17 provides the reference temperatures for the fuel basket and the MPC canister and, utilizing Tables 3.1.6 through 3.1.16, provides conservative numerical limits for the stresses and stress intensities for all loading cases. Reference temperatures for the MPC baseplate and the MPC lid are 400ºF

(204ºC) and 550ºF (288ºC), respectively, as specified in Table 2.2.3.

Finally, the lift devices in the HI-STAR 100 overpack and the multi-purpose canisters, collectively referred to as "trunnions", are subject to specific limits set forth by NUREG-0612: the primary stresses in a trunnion must be less than the smaller of 1/10 of the material ultimate strength and 1/6 of the material yield strength under a normal handling condition (Load Cases F2, E2, and 03 in Tables 3.1.3 through 3.1.5, respectively. The load combination D+H in Table 3.1.5 is equivalent to 1.15D. This is further explained in Subsection 3.4.3.

The region around the trunnions is part of the NF structure in HI-STAR 100 and an NB pressure boundary in the MPC, and as such, must satisfy the applicable stress (or stress intensity) limits for the load combination. In addition to meeting the applicable Code limits, it is further required that the local primary stresses at the trunnion/mother structure interface must not exceed the material yield stress at three times the handling condition load (1.15D). This criterion, mandated by Regulatory Guide 3.61, Section 3.4.3, eliminates the potential of local yielding at the trunnion/structure interface.

3.1.2.3 Brittle Fracture

The MPC canister and basket are constructed from a series of stainless steels termed Alloy X. These stainless steel materials do not undergo a ductile-to-brittle transition in the minimum temperature range of the HI-STAR 100 System. Therefore, brittle fracture is not a concern for the MPC components. However, the HI-STAR 100 overpack is composed of ferritic steel materials which will be subject to impact loading in a cold environment and, therefore, must be evaluated and/or subjected to impact testing in accordance with the ASME Code to ensure protection against brittle fracture.

Tables 3.1.18 and 3.1.19 provide the fracture toughness test criteria for the HI-STAR 100 components in accordance with the applicable ASME Code and Regulatory Guide requirements for prevention of brittle fracture.

All containment system materials subject to impact loading in a cold environment must be evaluated and/or tested for their potential for brittle fracture. Containment boundary plate and forging materials have been selected based on the material's capability to perform at low temperatures with excellent ductility properties. The lowest service temperature (LST) (where impactive or impulsive loads are present) is -20ºF (-29ºC) per Reg. Guide 7.8 [3.1.8]; however, HI-STAR 100 may be qualified to an LST of -40ºF (-40ºC). Table 3.1.18 provides the criteria for qualification to LST of either -20ºF (-29ºC) or LST of -40ºF (-40ºC). The appropriate Regulatory Guides are used to provide guidance on the test temperature to demonstrate appropriate resistance to brittle fracture. For component thicknesses greater than four-inches, such as cask bottom plate, top flange, and closure plate, the fracture arrest criteria of Regulatory Guide 7.12 [3.1.4] is used for component thicknesses up to 12 inches and a LST of -20ºF (-29ºC). In lieu of the fracture arrest criteria, the fracture initiation criteria from NUREG/CR-3826 [3.1.9] is used to determine the required Nil Ductility Transition (NDT) temperature, "T_{NDT}" where the component thickness is greater than 4 inches at an LST of -40ºF (-40ºC). As an alternative for components whose thickness exceeds 4 inches at an LST of -40ºF (-40ºC), the guidance from Regulatory Guide 7.12 may be followed by setting the required NDT temperature 20ºF (11.1ºC) below the NDT temperature applicable to an LST of -20ºF (-29ºC). For component thicknesses equal to or less than four-inches, such as the containment shell, the required Nil Ductility Transition (NDT) temperature, "TNDT" is determined based on the guidance from Regulatory Guide 7.11 [3.1.3] and NUREG/CR-1815 [3.1.5].

The closure plate bolts are fabricated from SB-637 Grade N07718, a high strength nickel alloy material. This material provides immunity from brittle fracture concerns at cask LST of either -29ºC (-20ºF) or LST of -40ºC (-40ºF) per Section III of the ASME code and NUREG/CR-1815.

ASME Code Section III, Subsection NF requires Charpy V-notch tests for materials of certain non-helium retention boundary components of the overpack. The intermediate shells used for gamma shielding are fabricated from normalized SA516-70. Table A1.15 of ASME Section IIA shows that normalized SA516-70 should have a minimum energy absorption of 12 ft-lb (16 N-m) at -40ºF (-40ºC) for a Charpy V-notch test. The lowest service temperature for the overpack is -40ºF (-40ºC). Therefore, these tests on the normalized SA516-70 materials of the intermediate shells will confirm the minimum energy absorption of 12 ft-lb (16 N-m) at -40ºF (-40ºC) and the ability of the intermediate shells to perform their intended function at the lowest service temperature.

The pocket trunnions (optional) are fabricated from 17-4PH material that is precipitation hardened to condition H1150. ARMCO Product Data Bulletin S-22 [Ref. 3.1.7] shows that Charpy V-notch testing of 17-4PH H1150 material at -110ºF (-79ºC) gives energy absorption values of approximately 48 ft-lb (65 N-m). This may be transferred into a fracture toughness value by using the relationship (presented in Section 4.2 of NUREG/CR-1815) between Charpy impact measurement, C_v (ft-lb) (N-m), and dynamic fracture toughness, K_{ID} (psi $\sqrt{\text{in.}}$)

$$K_{ID} = (5\ E\ C_v)^{1/2}$$

where E = 28.7 x 10^6 psi (197,880 MPa) and C_v (minimum) = 48 ft-lb (65 N-m). Therefore,

$$K_{ID} = 83 \text{ ksi } \sqrt{\text{in.}} \ (572 \text{ MPa } \sqrt{\text{mm}})$$

Using Figure 2 of NUREG/CR-1815 yields

$$T\text{-}T_{NDT} = 65^\circ F\ (18.3^\circ C)$$

and therefore

$$T_{NDT} = -110^\circ F - 65^\circ F = -175^\circ F\ (-115^\circ C)$$

While the pocket trunnions (optional) are not part of the helium retention boundary for the overpack, Regulatory Guide 7.12 is used to define the required T_{NDT} for the trunnion pocket thickness (T_{NDT} = -140ºF (-96ºC)). The 35ºF (1.7ºC) margin between the calculated T_{NDT} and the T_{NDT} defined in Regulatory Guide 7.12 provides assurance that brittle fracture failure of the 17-4PH material will not occur at the lowest service temperature.

3.1.2.4 Fatigue

In storage, the HI-STAR 100 System is not subject to significant cyclic loads. Failure due to fatigue is not a concern for the HI-STAR 100 System.

The system is subject to cyclic temperature fluctuations. These fluctuations result in small changes of thermal expansions and pressures in the MPC. The loads resulting from these changes are small and do not significantly contribute to the "usage factor" of the cask.

The closure plate bolts will be installed with a specified pre-load and, therefore, will be subject to little fluctuation in their state of stress due to small variations in overpack internal pressure.

Inspection of the trunnions specified in Chapter 9 will preclude use of a trunnion which exhibits visual damage.

3.1.2.5 Buckling

Certain load combinations subject structural sections with relatively large slenderness ratios (such as the enclosure vessel shell) to compressive stresses which may actuate buckling instability before the allowable stress is reached. Tables 3.1.4 and 3.1.5 list load combinations for the enclosure vessel and the HI-STAR 100 structure; the cases which warrant stability (buckling) check are listed therein.

Table 3.1.1

LOAD COMBINATIONS SIGNIFICANT TO HI-STAR 100 OVERPACK KINEMATIC STABILITY ANALYSIS

Load Case	**Combinations**†	**Comment**	**Analysis of this Load Case Presented in:**
A	D + F	This case establishes flood water flow velocity with a minimum safety factor of 1.1 against overturning and sliding.	Subsection 3.4.6
B	D + M + W'	Demonstrate that the HI-STAR 100 overpack with minimum SNF stored (minimum D) will not tip over.	Supplement 19 of [3.4.13]
C	D + E	Establish the value of ZPA†† which will not cause the overpack to tip over.	Subsection 3.4.7

† Loading symbols are defined in Table 2.2.13.
†† ZPA is zero period acceleration.

Table 3.1.2

DESIGN BASIS DECELERATIONS FOR THE DROP AND TIP-OVER EVENTS

Case	**Value (in multiples of acceleration due to gravity)**
Vertical axis drop	60
Horizontal axis (side) drop and tip-over	60

Table 3.1.3
LOADING CASES FOR THE FUEL BASKET

Load Case I.D.	**Loading**[†]	**Notes**	**Location Where this Case is Evaluated**
F1	T, T'	Demonstrate that the most adverse of the temperature distributions in the basket will not cause fuel basket to expand and contact the enclosure vessel wall.	Subsection 3.4.4.2
F2	D + H	For a lateral handling load, a 2g deceleration is imposed on the stored fuel.	Subsections 3.4.4.3.1.1, 3.4.4.4.1
F3			
F3.a	D + H'	Vertical axis drop event	Subsections 3.4.4.3.1.6, 3.4.4.3.1.3
F3.b	D + H'	Side Drop, 0° orientation (Figure 3.1.2)	Subsections 3.4.4.3.1.1, 3.4.4.4.1
F3.c	D + H'	Side Drop, 45° orientation (Figure 3.1.3)	Subsections 3.4.4.3.1.1, 3.4.4.4.1

† The symbols used for the loadings are defined in Table 2.2.13.

Table 3.1.4

LOADING CASES FOR THE ENCLOSURE VESSEL (CONFINEMENT BOUNDARY)

Load Case I.D.	Load Combination†	Notes	Comments and Location Where this Case is Analyzed
E1.a	Design internal pressure, P_i	Primary stress intensity limits in the shell, baseplate, and closure ring	Lid Supplement 14 of [3.4.14] Baseplate Supplement 15 of [3.4.14] Shell 3.4.4.3.1.2 Supports N/A
E1.b	Design external pressure, P_o	Primary stress intensity limits, buckling stability	Lid P_i bounds Baseplate P_i bounds Shell Supplement 23 of [3.4.13] Supports N/A
E1.c	Design internal pressure, P_i, plus Temperature, T	Primary plus secondary stress intensity under Level A condition	Subsection 3.4.4.3.1.2
E2	D + H + (P_i, P_o)†† For elastic stability, only D+H is considered.	Vertical lift, internal operating pressure conservatively assumed to be equal to the normal design pressure.	Lid Supplement 14 of [3.4.14] Baseplate Supplement 15 of [3.4.14] Shell 3.4.4.3.1.1, 3.4.4.4.1 (stress) Supplement 23 of [3.4.13] Supports 3.4.4.3.1.1, 3.4.4.4.1

† The symbols used for the loadings are defined in Table 2.2.13.

†† The notation (P_i, P_o) means that both cases are checked for stresses with either P_o or P_i applied.

Table 3.1.4 (continued)

LOADING CASES FOR THE ENCLOSURE VESSEL (CONFINEMENT BOUNDARY)

Load Case I.D.	Load Combination†	Notes	Comments and Location Where this Case is Analyzed
E3			
E3.a	D + H' + (P_o, P_i) For elastic stability, only D+H' is considered.	Vertical axis drop event	Lid Supplement 14 of [3.4.14] Baseplate Supplement 15 of [3.4.14] Shell Supplement 23 of [3.4.13] Supports N/A
E3.b	D + H' + (P_i, P_o)	Side drop, 0° orientation (Figure 3.1.2)	Lid End drop bounds Baseplate End drop bounds Shell 3.4.4.3.1.1, 3.4.4.4.1 Supports 3.4.4.3.1.1, 3.4.4.4.1, Supplement 18 of [3.4.14]
E3.c	D + H' + (P_i, P_o)	Side drop, 45° orientation (Figure 3.1.3)	Lid End drop bounds Baseplate End drop bounds Shell 3.4.4.3.1.1, 3.4.4.4.1 Supports 3.4.4.3.1.1, 3.4.4.4.1, Supplement 18 of [3.4.14]

Table 3.1.4 (continued)

LOADING CASES FOR THE ENCLOSURE VESSEL (CONFINEMENT BOUNDARY)

Load Case I.D.	Load Combination†	Notes	Comments and Location Where this Case is Analyzed
E4	T	Demonstrate that interference with the overpack will not develop for T.	Subsection 3.4.4.2
E5	P_i^* or P_o^* + D + T^*	Demonstrate compliance with Level D stress limits - buckling stability.	Lid: Supplements 14 and 28 of [3.4.14] Baseplate: Supplements 15 and 28 of [3.4.14] Shell: Supplement 23 of [3.4.13] 3.4.4.3.1.5 (thermal stress) Supports: N/A

Table 3.1.5
LOAD CASES FOR THE HI-STAR 100 OVERPACK

Load Case I.D.	Loading†	Notes	Location in FSAR or Calc. Package Where this Case is Analyzed
01	(P_i, P_o)	Compliance with NB stress intensity limits	3.4.4.3.2.1, 3.4.4.4.2
02	(P_i^*, P_o^*) + D + T*	Compliance with NB Level D stress intensity limits	3.4.4.3.2.1, 3.4.4.4.2
03	D + H + T + (P_o, P_i)	Vertical load handling of HI-STAR 100 Overpack	3.4.4.3.2.1, 3.4.4.4.2; Supplement 20 of [3.4.13]
04			
04.a	D + H' + (P_o, P_i)	End drop; primary stress intensities must meet Level D limits.	3.4.4.3.2.1, 3.4.4.4.2
04.b	D + H' + (P_o, P_i)	Horizontal (side) drop; meet Level D limits for NF components away from the impacted zone	3.4.4.3.2.1, 3.4.4.4.2
05	T	Satisfy primary membrane plus bending stress limits for NB components	3.4.4.3.2.1, 3.4.4.4.2
06	M (small and medium penetrant missiles)	Demonstrate that no thru-wall breach of the overpack occurs, no loss of helium retention boundary occurs, and that primary stress levels are not exceeded.	Supplement 22 of [3.4.13]

Notes:

1. Under each of these load cases, different regions of the structure are analyzed to demonstrate compliance.

† The symbols used for the loadings are defined in Table 2.2.13.

Table 3.1.6

DESIGN, LEVELS A AND B: STRESS INTENSITY

Code: ASME NB
Material: SA203-E
Service Conditions: Design, Levels A and B
Item: Stress Intensity

Temp. (ºF) (ºC)	Classification and Value (ksi) (MPa)					
	S_m	P_m†	P_L†	$P_L + P_b$†	$P_L + P_b + Q$††	P_e††
-20 to 100 (-29 to 38)	23.3 (160.6)	23.3 (160.6)	35.0 (241.3)	35.0 (241.3)	69.9 (481.9)	69.9 (481.9)
200 (93)	23.3 (160.6)	23.3 (160.6)	35.0 (241.3)	35.0 (241.3)	69.9 (481.9)	69.9 (481.9)
300 (149)	23.3 (160.6)	23.3 (160.6)	35.0 (241.3)	35.0 (241.3)	69.9 (481.9)	69.9 (481.9)
400 (204)	22.9 (157.9)	22.9 (157.9)	34.4 (237.2)	34.4 (237.2)	68.7 (473.7)	68.7 (473.7)
500 (260)	21.6 (148.9)	21.6 (148.9)	32.4 (223.4)	32.4 (223.4)	64.8 (446.8)	64.8 (446.8)

Definitions:

S_m = Stress intensity values per ASME Code
P_m = Primary membrane stress intensity
P_L = Local membrane stress intensity
P_b = Primary bending stress intensity
P_e = Expansion stress
Q = Secondary stress
P_L+P_b = Either primary or local membrane plus primary bending

Definitions for Table 3.1.6 apply to all following tables unless modified.

Notes:

1. Limits on values are presented in Table 2.2.10.

† Evaluation required for Design condition only.
†† Evaluation required for Levels A and B conditions only. P_e not applicable to vessels.

Table 3.1.7

LEVEL D: STRESS INTENSITY

Code: ASME NB
Material: SA203-E
Service Condition: Level D
Item: Stress Intensity

Temp. (ºF) (ºC)	Classification and Value (ksi) (MPa)		
	P_m	P_L	$P_L + P_b$
-20 to 100 (-29 to 38)	49.0 (337.8)	70.0 (482.6)	70.0 (482.6)
200 (93)	49.0 (337.8)	70.0 (482.6)	70.0 (482.6)
300 (149)	49.0 (337.8)	70.0 (482.6)	70.0 (482.6)
400 (204)	48.2 (332.3)	68.8 (474.4)	68.8 (474.4)
500 (260)	45.4 (313.0)	64.9 (447.5)	64.9 (447.5)

Notes:

1. Level D allowables per NB-3225 and Appendix F, Paragraph F-1331.

2. Average primary shear stress across a section loaded in pure shear may not exceed 0.42 S_u.

3. Limits on values are presented in Table 2.2.10.

4. P_m, P_L, and P_b are defined in Table 3.1.6.

Table 3.1.8

DESIGN, LEVELS A AND B: STRESS INTENSITY

Code: ASME NB
Material: SA350-LF3
Service Conditions Design, Levels A and B
Item: Stress Intensity

Temp. (°F) (°C)	Classification and Value (ksi) (MPa)					
	S_m	P_m†	P_L†	$P_L + P_b$†	$P_L + P_b$ +Q††	P_e††
-20 to 100 (-29 to 38)	23.3 (160.6)	23.3 (160.6)	35.0 (241.3)	35.0 (241.3)	69.9 (481.9)	69.9 (481.9)
200 (93)	22.8 (157.2)	22.8 (157.2)	34.2 (235.8)	34.2 (235.8)	68.4 (471.6)	68.4 (471.6)
300 (149)	22.2 (153.1)	22.2 (153.1)	33.3 (229.6)	33.3 (229.6)	66.6 (459.2)	66.6 (459.2)
400 (204)	21.5 (148.2)	21.5 (148.2)	32.3 (222.7)	32.3 (222.7)	64.5 (444.7)	64.5 (444.7)
500 (260)	20.2 (139.3)	20.2 (139.3)	30.3 (208.9)	30.3 (208.9)	60.6 (417.8)	60.6 (417.8)
600 (316)	18.5 (127.6)	18.5 (127.6)	27.75 (191.3)	27.75 (191.3)	55.5 (382.6)	55.5 (382.6)
700 (371)	16.8 (115.8)	16.8 (115.8)	25.2 (173.7)	25.2 (173.7)	50.4 (347.5)	50.4 (347.5)

Notes:

1. Source for S_m is ASME Code.
2. Limits on values are presented in Table 2.2.10.
3. S_m, P_m, P_L, P_b, Q, and P_e are defined in Table 3.1.6.

† Evaluation required for Design condition only.
†† Evaluation required for Levels A and B conditions only. P_e not applicable to vessels.

Table 3.1.9

LEVEL D, STRESS INTENSITY

Code: ASME NB
Material: SA350-LF3
Service Conditions: Level D
Item: Stress Intensity

Temp. (ºF) (ºC)	Classification and Value (ksi) (MPa)		
	P_m	P_L	$P_L + P_b$
-20 to 100 (-29 to 38)	49.0 (337.8)	70.0 (482.6)	70.0 (482.6)
200 (93)	48.0 (330.9)	68.5 (472.3)	68.5 (472.3)
300 (149)	46.7 (322.0)	66.7 (459.9)	66.7 (459.9)
400 (204)	45.2 (311.6)	64.6 (455.4)	64.6 (455.4)
500 (260)	42.5 (293.0)	60.7 (418.5)	60.7 (418.5)
600 (316)	38.9 (268.2)	58.4 (402.6)	58.4 (402.6)
700 (371)	35.3 (243.4)	53.1 (366.1)	53.1 (366.1)

Notes:

1. Level D allowables per NB-3225 and Appendix F, Paragraph F-1331.
2. Average primary shear stress across a section loaded in pure shear may not exceed 0.42 S_u.
3. Limits on values are presented in Table 2.2.10.
4. P_m, P_L, and P_b are defined in Table 3.1.6.

Table 3.1.10

DESIGN AND LEVEL A: STRESS

Code: ASME NF
Material: SA516, Grade 70, SA515, Grade 70
Service Conditions: Design and Level A
Item: Stress

Temp. (^{O}F) (^{O}C)	Classification and Value (ksi) (MPa)		
	S	Membrane Stress	Membrane plus Bending Stress
-20 to 650 (-29 to 343)	17.5 (120.6)	17.5 (120.6)	26.3 (181.3)
700 (371)	16.6 (114.4)	16.6 (114.4)	24.9 (171.7)

Notes:

1. S = Maximum allowable stress values from Table 1A of ASME Code, Section II, Part D.
2. Stress classification per Paragraph NF-3260.
3. Limits on values are presented in Table 2.2.12.

Table 3.1.11

LEVEL B: STRESS

Code: ASME NF
Material: SA516, Grade 70, SA515, Grade 70
Service Conditions: Level B
Item: Stress

Temp. (^{O}F) (^{O}C)	Classification and Value (ksi) (MPa)	
	Membrane Stress	Membrane plus Bending Stress
-20 to 650 (-29 to 343)	23.3 (160.6)	34.9 (240.6)
700 (371)	22.1 (152.4)	33.1 (228.2)

Notes:

1. Limits on values are presented in Table 2.2.12 with allowables from Table 3.1.10.

Table 3.1.12

LEVEL D: STRESS INTENSITY

Code: ASME NF
Material: SA516, Grade 70, SA515, Grade 70
Service Conditions: Level D
Item: Stress Intensity

Temp. (^{O}F) (^{O}C)	Classification and Value (ksi) (MPa)		
	S_m	P_m	$P_m + P_b$
-20 to 100 (-29 to 38)	23.3 (160.6)	45.6 (314.4)	68.4 (471.6)
200 (93)	23.1 (159.3)	41.5 (286.1)	62.3 (429.5)
300 (149)	22.5 (155.1)	40.4 (278.5)	60.6 (417.8)
400 (204)	21.7 (149.6)	39.1 (269.6)	58.7 (404.7)
500 (260)	20.5 (141.3)	36.8 (253.7)	55.3 (381.3)
600 (316)	18.7 (128.9)	33.7 (232.4)	50.6 (348.9)
650 (343)	18.4 (126.9)	33.1 (228.2)	49.7 (342.7)
700 (371)	18.3 (126.2)	32.9 (226.8)	49.3 (339.9)

Notes:

1. Level D allowable stress intensities per Appendix F, Paragraph F-1332.

2. S_m = Stress intensity values per Table 2A of ASME, Section II, Part D.

3. Limits on values are presented in Table 2.2.12.

4. P_m and P_b are defined in Table 3.1.6.

Table 3.1.13

DESIGN, LEVELS A AND B: STRESS INTENSITY

Code: ASME NB
Material: Alloy X
Service Conditions: Design, Levels A and B
Item: Stress Intensity

Temp. (°F) (°C)	Classification and Numerical Value (ksi) (MPa)					
	S_m	P_m†	P_L†	$P_L + P_b$†	P_L+P_b+Q††	P_e††
-20 to 100 (-29 to 38)	20.0 (137.9)	20.0 (137.9)	30.0 (206.8)	30.0 (206.8)	60.0 (413.7)	60.0 (413.7)
200 (93)	20.0 (137.9)	20.0 (137.9)	30.0 (206.8)	30.0 (206.8)	60.0 (413.7)	60.0 (413.7)
300 (149)	20.0 (137.9)	20.0 (137.9)	30.0 (206.8)	30.0 (206.8)	60.0 (413.7)	60.0 (413.7)
400 (204)	18.7 (128.9)	18.7 (128.9)	28.1 (193.7)	28.1 (193.7)	56.1 (386.8)	56.1 (386.8)
500 (260)	17.5 (120.6)	17.5 (120.6)	26.3 (181.3)	26.3 (181.3)	52.5 (362.0)	52.5 (362.0)
600 (316)	16.4 (113.1)	16.4 (113.1)	24.6 (169.6)	24.6 (169.6)	49.2 (339.2)	49.2 (339.2)
650 (343)	16.0 (110.3)	16.0 (110.3)	24.0 (165.5)	24.0 (165.5)	48.0 (330.9)	48.0 (330.9)
700 (371)	15.6 (107.6)	15.6 (107.6)	23.4 (161.3)	23.4 (161.3)	46.8 (322.7)	46.8 (322.7)
750 (399)	15.2 (104.8)	15.2 (104.8)	22.8 (157.2)	22.8 (157.2)	45.6 (314.4)	45.6 (314.4)
800 (427)	14.9 (102.7)	14.9 (102.7)	22.4 (154.4)	22.4 (154.4)	44.7 (308.2)	44.7 (308.2)

Notes:

1. S_m = Stress intensity values per Table 2A of ASME II, Part D.
2. Alloy X S_m values are the lowest values for each of the candidate materials at temperature.
3. Stress classification per NB-3220.
4. Limits on values are presented in Table 2.2.10.
5. P_m, P_L, P_b, Q, and P_e are defined in Table 3.1.6.

† Evaluation required for Design condition only.
†† Evaluation required for Levels A, B conditions only. P_e not applicable to vessels.

Table 3.1.14

LEVEL D: STRESS INTENSITY

Code: ASME NB
Material: Alloy X
Service Conditions: Level D
Item: Stress Intensity

Temp. (^{O}F) (^{O}C)	Classification and Value (ksi) (MPa)		
	P_m	P_L	$P_L + P_b$
-20 to 100 (-29 to 38)	48.0 (330.9)	72.0 (496.4)	72.0 (496.4)
200 (93)	48.0 (330.9)	72.0 (494.4)	72.0 (496.4)
300 (149)	46.2 (318.5)	69.3 (477.8)	69.3 (477.8)
400 (204)	44.9 (309.6)	67.4 (464.7)	67.4 (464.7)
500 (260)	42.0 (289.6)	63.0 (434.4)	63.0 (434.4)
600 (316)	39.4 (271.6)	59.1 (407.5)	59.1 (407.5)
650 (343)	38.4 (264.8)	57.6 (397.1)	57.6 (397.1)
700 (371)	37.4 (257.9)	56.1 (386.8)	56.1 (386.8)
750 (399)	36.5 (251.6)	54.8 (377.8)	54.8 (377.8)
800 (427)	35.8 (246.8)	53.7 (370.2)	53.7 (370.2)

Notes:

1. Level D stress intensities per ASME NB-3225 and Appendix F, Paragraph F-1331.

2. The average primary shear strength across a section loaded in pure shear may not exceed 0.42 S_u.

3. Limits on values are presented in Table 2.2.10.

4. P_m, P_L, and P_b are defined in Table 3.1.6.

Table 3.1.15

DESIGN, LEVELS A AND B: STRESS INTENSITY

Code: ASME NG
Material: Alloy X
Service Conditions: Design, Levels A and B
Item: Stress Intensity

Temp. (^{O}F) (^{O}C)	Classification and Value (ksi) (MPa)				
	S_m	P_m	P_m+P_b	P_m+P_b+Q	P_e
-20 to 100 (-29 to 38)	20.0 (137.9)	20.0 (137.9)	30.0 (206.8)	60.0 (413.7)	60.0 (413.7)
200 (93)	20.0 (137.9)	20.0 (137.9)	30.0 (206.8)	60.0 (413.7)	60.0 (413.7)
300 (149)	20.0 (137.9)	20.0 (137.9)	30.0 (206.8)	60.0 (413.7)	60.0 (413.7)
400 (204)	18.7 (128.9)	18.7 (128.9)	28.1 (193.7)	56.1 (386.8)	56.1 (386.8)
500 (260)	17.5 (120.6)	17.5 (120.6)	26.3 (181.3)	52.5 (362.0)	52.5 (362.0)
600 (316)	16.4 (113.1)	16.4 (113.1)	24.6 (169.6)	49.2 (339.2)	49.2 (339.2)
650 (343)	16.0 (110.3)	16.0 (110.3)	24.0 (165.5)	48.0 (330.9)	48.0 (330.9)
700 (371)	15.6 (107.6)	15.6 (107.6)	23.4 (161.3)	46.8 (322.7)	46.8 (322.7)
750 (399)	15.2 (104.8)	15.2 (104.8)	22.8 (157.2)	45.6 (314.4)	45.6 (314.4)
800 (427)	14.9 (102.7)	14.9 (102.7)	22.4 (154.4)	44.7 (308.2)	44.7 (308.2)

Notes:

1. S_m = Stress intensity values per Table 2A of ASME, Section II, Part D.

2. Alloy X S_m values are the lowest values for each of the candidate materials at temperature.

3. Classifications per NG-3220.

4. Limits on values are presented in Table 2.2.11.

5. P_m, P_b, Q, and P_e are defined in Table 3.1.6.

Table 3.1.16

LEVEL D: STRESS INTENSITY

Code: ASME NG
Material: Alloy X
Service Conditions: Level D
Item: Stress Intensity

Temp. (°F) (°C)	Classification and Value (ksi) (MPa)		
	P_m	P_L	$P_L + P_b$
-20 to 100 (-29 to 38)	48.0 (330.9)	72.0 (496.4)	72.0 (496.4)
200 (93)	48.0 (330.9)	72.0 (494.4)	72.0 (496.4)
300 (149)	46.2 (318.5)	69.3 (477.8)	69.3 (477.8)
400 (204)	44.9 (309.6)	67.4 (464.7)	67.4 (464.7)
500 (260)	42.0 (289.6)	63.0 (434.4)	63.0 (434.4)
600 (316)	39.4 (271.6)	59.1 (407.5)	59.1 (407.5)
650 (343)	38.4 (264.8)	57.6 (397.1)	57.6 (397.1)
700 (371)	37.4 (257.9)	56.1 (386.8)	56.1 (386.8)
750 (399)	36.5 (251.6)	54.8 (377.8)	54.8 (377.8)
800 (427)	35.8 (246.8)	53.7 (370.2)	53.7 (370.2)

Notes:

1. Level D stress intensities per ASME NG-3225 and Appendix F, Paragraph F-1331.

2. The average primary shear strength across a section loaded in pure shear may not exceed 0.42 S_u.

3. Limits on values are presented in Table 2.2.11.

4. P_m, P_L, and P_b are defined in Table 3.1.6.

Table 3.1.17

REFERENCE TEMPERATURES AND STRESS LIMITS
FOR THE VARIOUS LOAD CASES

Load Case I.D.	Material	Reference Temperature†, °F (°C)	Stress Intensity Allowables, ksi (MPa)		
			P_m	$P_L + P_b$	$P_L + P_b + Q$
F1	Alloy X	725 (385)	15.4 (106.2)	23.1 (159.3)	46.2 (318.5)
F2	Alloy X	725 (385)	15.4 (106.2)	23.1 (159.3)	46.2 (318.5)
F3	Alloy X	725 (385)	36.9 (254.4)	55.4 (382.0)	NL††
E1	Alloy X	450 (232)	18.1 (124.8)	27.2 (187.5)	54.3 (374.4)
E2	Alloy X	450 (232)	18.1 (124.8)	27.2 (187.5)	54.3 (374.4)
E3	Alloy X	450 (232)	43.4 (299.2)	65.2 (449.5)	NL
E4	Alloy X	450 (232)	18.1 (124.8)	27.2 (187.5)	54.3 (374.4)
E5	Alloy X	775 (413)	36.15 (249.2)	54.25 (374.04)	NL

Note:

1. Q, P_m, P_L, and P_b are defined in Table 3.1.6.

† Values for reference temperatures are taken as the design temperatures (Table 2.2.3).
†† NL: No specified limit in the Code.

Table 3.1.17 (continued):
REFERENCE TEMPERATURES AND STRESS LIMITS
FOR THE VARIOUS LOAD CASES

Load Case I.D.	Material	Reference Temperature,†, †† °F (°C)	Stress Intensity Allowables, ksi (MPa)		
			P_m	$P_L + P_b$	$P_L + P_b + Q$
O1	SA203-E	400 (204)	22.9 (157.9)	34.4 (237.2)	68.7 (473.7)
	SA350-LF3	400 (204)	21.5 (148.2)	32.3 (222.7)	64.5 (444.7)
	SA516 Gr. 70 SA515 Gr. 70	400 (204)	17.5 (120.6)	26.3 (181.3)	NL†††
O2	SA203-E	400 (204)	48.2 (332.3)	68.8 (474.4)	NL
	SA350-LF3	400 (204)	45.2 (311.6)	64.6 (445.4)	NL
	SA516 Gr. 70 SA515 Gr. 70	400 (204)	39.1 (269.6)	58.7 (404.7)	NL
O3	SA203-E	400 (204)	22.9 (157.9)	34.4 (237.2)	68.7 (473.7)
	SA350-LF3	400 (204)	21.5 (148.2)	32.3 (222.7)	64.5 (444.7)
	SA516 Gr. 70 SA515 Gr. 70	400 (204)	17.5 (120.6)	26.3 (181.3)	NL
O4	SA203-E	400 (204)	48.2 (332.3)	68.8 (474.4)	NL
	SA350-LF3	400 (204)	45.2 (311.6)	64.6 (445.4)	NL
	SA516 Gr. 70 SA515 Gr. 70	400 (204)	39.1 (269.6)	58.7 (404.7)	NL
O5	SA203-E	400 (204)	22.9 (157.9)	34.4 (237.2)	68.7 (473.7)
	SA350-LF3	400 (204)	21.5 (148.2)	32.3 (222.7)	64.5 (444.7)
	SA516 Gr. 70 SA515 Gr. 70	400 (204)	17.5 (120.6)	26.3 (181.3)	NL
O6	SA203-E	400 (204)	48.2 (332.3)	68.8 (474.4)	NL
	SA350-LF3	400 (204)	45.2 (311.6)	64.6 (445.4)	NL
	SA516 Gr. 70 SA515 Gr. 70	400 (204)	39.1 (269.6)	58.7 (404.7)	NL

Note: 1. P_m, P_L, P_b, and Q are defined in Table 3.1.6.

† Values for reference temperatures are taken as the design temperatures (Table 2.2.3).
†† For storage fire analysis, temperatures are defined by thermal solution.
††† NL: No specified limit in the Code.

Table 3.1.18
FRACTURE TOUGHNESS TEST CRITERIA: HELIUM RETENTION BOUNDARY (Sheet 1 of 2)

Item	Material	Thickness in (mm)	Qualification to LST of -29°C (-20°F) (Note 3)		Qualification to LST of -40°C (-40°F) (Note 3)	
			Charpy V-Notch Temperature	**Drop Test Temperature (Note 1)**	**Charpy V-Notch Temperature**	**Drop Test Temperature (Note 1)**
Weld Metal for NB Welds	As required	NA	$T_{NDT} \leq$ -102.5°F (-74.7°C) with testing and acceptance criteria per ASME Section III, Subsection NB, Article NB-2430 and Article NB-2330	$T_{NDT} \leq$ -102.5°F (-74.7°C) based on containment shell thickness	$T_{NDT} \leq$ -122.5°F (-85.8°C) with testing and acceptance criteria per ASME Section III, Subsection NB, Article NB-2430 and Article NB-2330	$T_{NDT} \leq$ -122.5°F (-85.8°C) based on containment shell thickness
Inner Shell	SA-203 E/ SA-350 LF3	2.5 (64)	$T_{NDT} \leq$ -102.5°F (-74.7°C) with testing and acceptance criteria per ASME Section III, Subsection NB, Article NB-2330	$T_{NDT} \leq$ -102.5°F (-74.7°C) per R.G. 7.11	$T_{NDT} \leq$ -122.5°F (-85.8°C) with testing and acceptance criteria per ASME Section III, Subsection NB, Article NB-2330	$T_{NDT} \leq$ -122.5°F (-85.8°C) per R.G. 7.11
Top Flange	SA-350 LF3	8.75 (222)	$T_{NDT} \leq$ -135.9°F (-93.3°C) with testing and acceptance criteria per ASME Section III, Subsection NB, Article NB-2330	$T_{NDT} \leq$ -135.9°F (-93.3°C) per R.G 7.12.	$T_{NDT} \leq$ -112.8°F (-80.4°C) with testing and acceptance criteria per ASME Section III, Subsection NB, Article NB-2330	$T_{NDT} \leq$ -112.8°F (-80.4°C) per fracture initiation criteria developed in the NUREG-CR-3826 (Notes 4 and 5)
Bottom Plate	SA-350 LF3	6 (152)	$T_{NDT} \leq$ -129°F (-89.4°C) with testing and acceptance criteria per ASME Section III, Subsection NB, Article NB-2330	$T_{NDT} \leq$ -129°F (-89.4°C) per R.G 7.12.	$T_{NDT} \leq$ -97°F (-71.7°C) with testing and acceptance criteria per ASME Section III, Subsection NB, Article NB-2330	$T_{NDT} \leq$ -97°F (-71.7°C) per fracture initiation criteria developed in the NUREG-CR-3826 (Notes 4 and 5)

Table 3.1.18
FRACTURE TOUGHNESS TEST CRITERIA: HELIUM RETENTION BOUNDARY (Sheet 2 of 2)

Item	Material	Thickness in (mm)	Qualification to LST of -29°C (-20°F) (Note 3)		Qualification to LST of -40°C (-40°F) (Note 3)	
			Charpy V-Notch Temperature	**Drop Test Temperature (Note 1)**	**Charpy V-Notch Temperature**	**Drop Test Temperature (Note 1)**
Closure Plate	SA-350 LF3	6 (152)	$T_{NDT} \leq$ -129ºF (-89.4ºC) with testing and acceptance criteria per ASME Section III, Subsection NB, Article NB-2330	$T_{NDT} \leq$ -129ºF (-89.4ºC) per R.G 7.12.	$T_{NDT} \leq$ -97ºF (-71.7ºC) with testing and acceptance criteria per ASME Section III, Subsection NB, Article NB-2330	$T_{NDT} \leq$ -97ºF (-71.7ºC) per fracture initiation criteria developed in the NUREG-CR-3826 (Notes 4 and 5)

Notes:

1. Materials to be tested in accordance with ASTM E208-87a.
2. An additional Charpy absorbed energy requirement of 5 ft-lb at -29ºC (-20ºF) or at -40⁰C (-40⁰F), depending on the desired cask LST qualification, is imposed on the closure lid bolts.
3. Per Reg. Guide 7.8 [3.1.8], the LST which applies to impactive loads is -29⁰C (-20⁰F). The cask may be qualified to either to an LST of either -29⁰C (-20⁰F) or -40⁰C (-40⁰F).
4. Component to undergo 100% volumetric examination to confirm absence of flaws which exceed the critical values as defined in NUREG/CR-3826 Table 3. 100% volumetric re-examination is required for cask components qualified per NUREG/CR-3826 following cask operations which result in impactive or impulsive loadings in excess of those defined in the normal conditions of transport.
5. In lieu of qualification per NUREG/CR-3826, qualification per R.G. 7.12 may be applied.

Table 3.1.19
FRACTURE TOUGHNESS TEST CRITERIA
MISCELLANEOUS ITEMS

Item	Material	Thickness (in) (mm)	Charpy V-Notch Temperature†	Drop Test Temperature
Intermediate Shells	SA516 Grade 70	1-1/4 and 1 (31.75 and 25.4)	Test temperature = -409ºF (-245ºC) with acceptance criteria per ASME Section III, Subsection NF, Table NF-2331(a)-3 and Figure NF-2331(a)-2	Not Required
Port Cover Plates	SA203-E	1-1/2 (38.1)	Test temperature = -40ºF (-40ºC) with acceptance criteria per ASME Section III, Subsection NF, Table NF--2331(a)-3 and Figure NF-2331(a)-2	Not Required
Weld Metal for NF Welds	As required	NA	As required per ASME Section III, Subsection NF, Article NF-2430 and Article NF-2330 Test temperature = -40ºF (-40ºC)	Not Required

† Temperature is T_{NDT} unless noted.

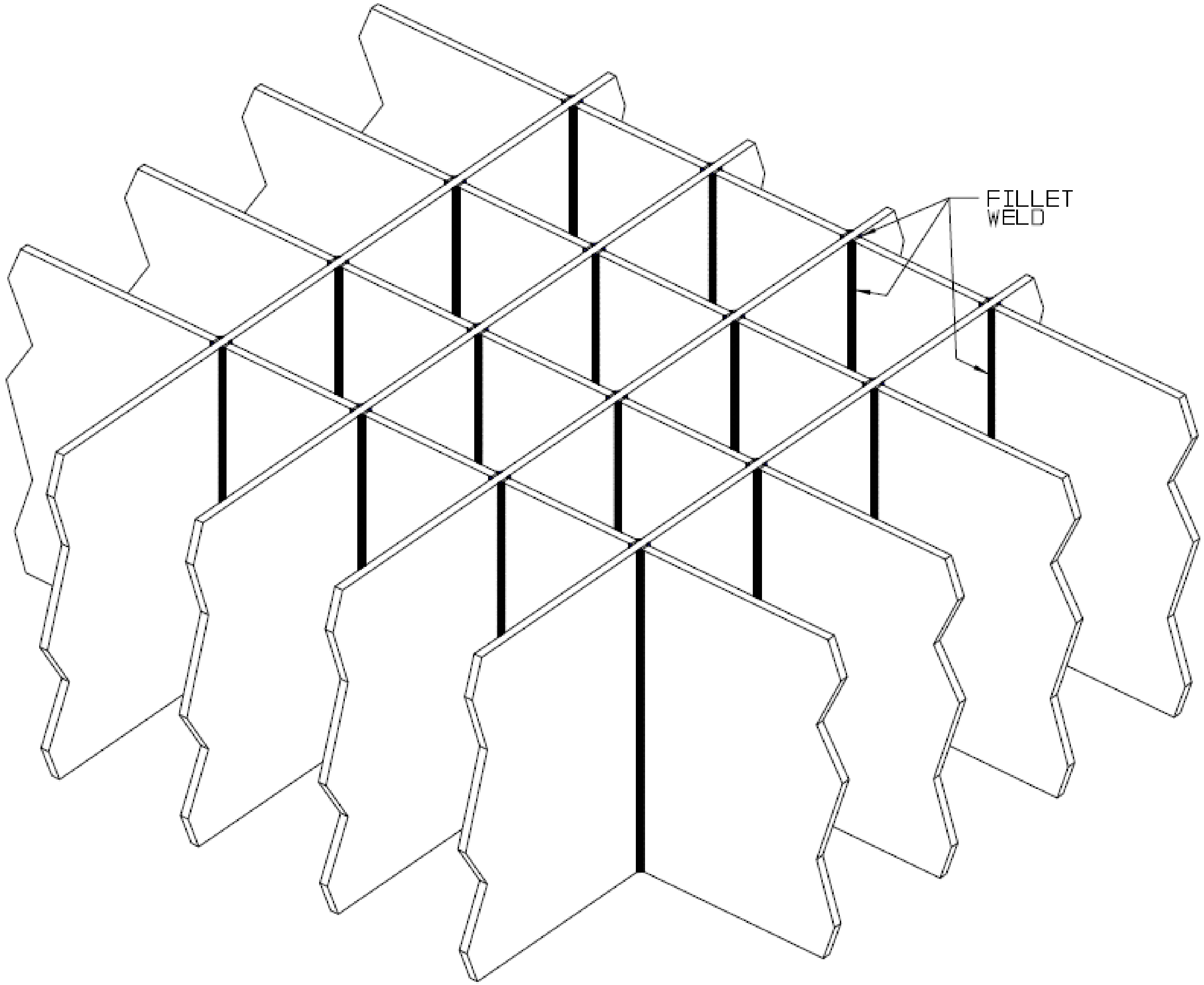

FIGURE 3.1.1; MPC FUEL BASKET GEOMETRY

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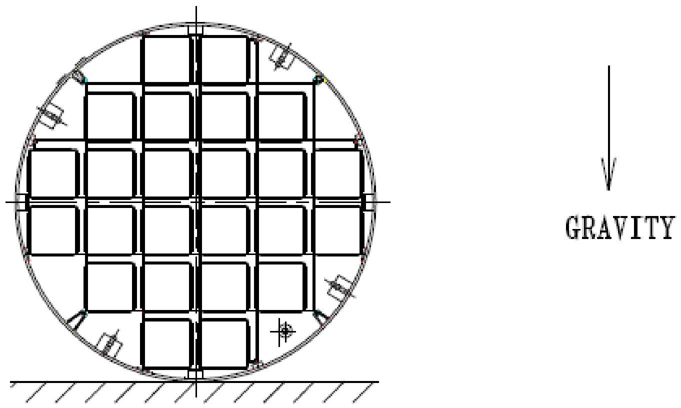

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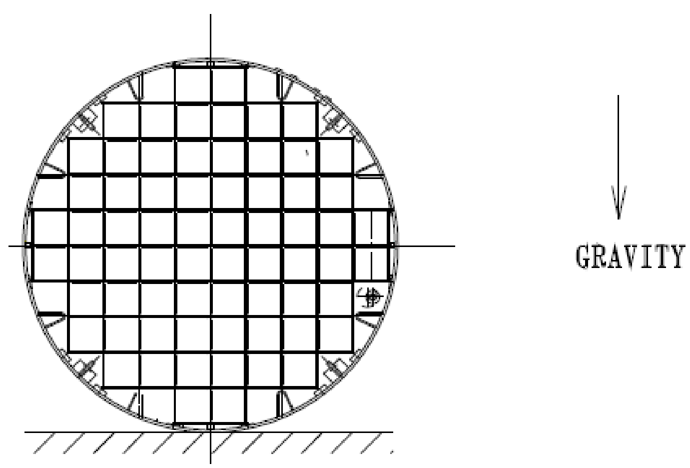

FIGURE 3.1.2; 0° DROP ORIENTATIONS FOR THE MPCs

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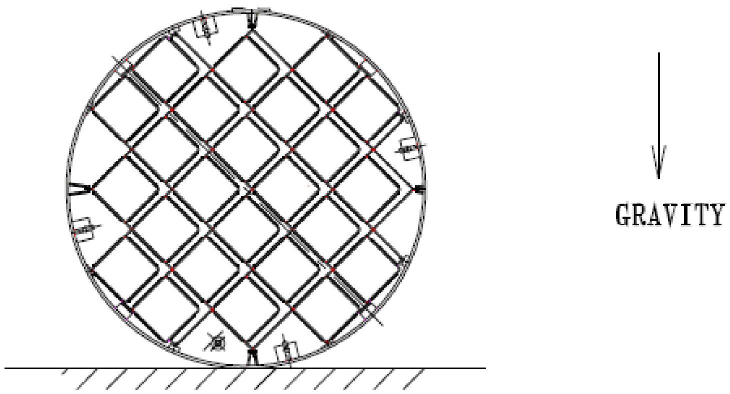

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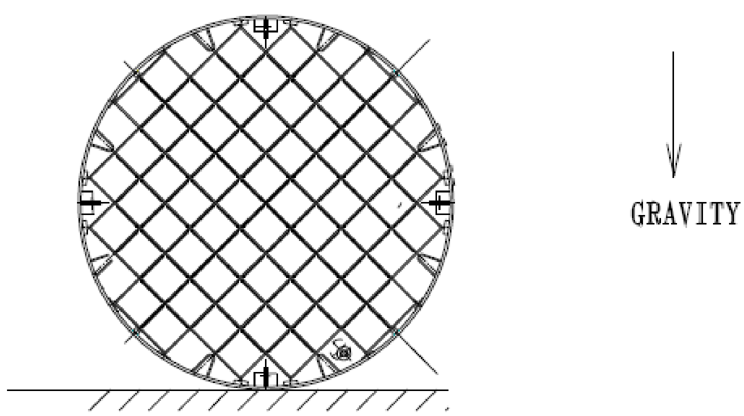

FIGURE 3.1.3; 45° DROP ORIENTATIONS FOR THE MPCs

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3.2 WEIGHTS AND CENTERS OF GRAVITY

Table 3.2.1 provides the weights of the individual HI-STAR 100 components as well as the total system weights. Contained water during loading is not included in this table.

The locations of the calculated centers of gravity (CGs) are presented in Table 3.2.2. All centers of gravity are located on the cask centerline, since the non-axisymmetry effects of the cask system plus contents are negligible.

Table 3.2.3 provides the lift weight when the HI-STAR 100 System with the heaviest fully loaded MPC is being lifted from the fuel pool. The effect of buoyancy is neglected, and the weight of rigging is set at a conservative value.

Table 3.2.4 provides a set of bounding weights that may be used in analytical calculations.

Table 3.2.1

HI-STAR 100 WEIGHT DATA[†]

	CALCULATED WEIGHT (lb) (kN)	
Item	**Component**	**Assembly**
• Overpack • Overpack closure plate	 7,984 (35.5)	 153,710 (683.7)
• MPC-24 • Fuel basket • Without SNF • Fully loaded with SNF	 20,842 (92.7)	 40,868 (181.8) 82,494 (367.0)
• Overpack with loaded MPC-24		236,204 (1,051)
• MPC-68 • Fuel basket • Without SNF • Fully loaded with SNF	 16,240 (72.2)	 37,591 (167.2) 87,171 (387.8)
• Overpack with fully loaded MPC-68		240,881 (1,071)
• MPC-32[1] • Fuel basket • Without SNF • Fully loaded with SNF	 12,340 (54.9)	 34,507 (153.5) 89,765 (399.3)
• Overpack with fully loaded MPC-32		243,475 (1,083)
• Overpack with minimum weight MPC without SNF (Value listed is lower bound to actual minimum weight of 191,301 lb (850.9 kN))		189,000 (840.7)

† All calculated weights are rounded to the nearest pound

1 The data for MPC-32 is adapted from Table 2.2.1 of [1.2.6]

Table 3.2.2

CENTERS OF GRAVITY OF HI-STAR 100 CONFIGURATIONS

Component	Height of CG Above Datum†, inches (mm)
Overpack empty	99.7 (2,532)
MPC-24 empty	109.0 (2,769)
MPC-68 empty	111.5 (2,832)
MPC-32 empty[1]	113.2 (2,875)
MPC-24 with fuel in overpack	102.9 (2,614)
MPC-68 with fuel in overpack	103.2 (2,621)
MPC-32 with fuel in overpack	102.1 (2,593)

† The datum used for calculations involving the overpack is the bottom of the overpack bottom plate. The datum used for calculations involving the MPC only is the bottom of MPC baseplate (see Figure 3.2.1).

1 The data for MPC-32 is adapted from Table 2.2.2 of [1.2.6].

Table 3.2.3

LIFT WEIGHT ABOVE POOL

Item	**Calculated Weight (lb)† (kN)**
Total weight of overpack	153,710 (683.7)
Total weight of an MPC (Upper Bound) + fuel	89,765†† (399.3)
Overpack closure plate	-7,984 (-35.5)
Water in MPC and overpack	16,384 (72.9)
Lift yoke	3,600 (16.0)
Inflatable annulus seal	50 (0.2)
TOTAL	255,525††† (1,137)

† The actual weight of some of these items may vary in the field due to differences in client procedures for performing loading/unloading operations.

†† Includes MPC closure ring.

††† Trunnion rating and crane limits at certain sites may require temporary water removal from the HI-STAR 100 System during removal from the pool (See Chapter 8).

Table 3.2.4
COMPONENT WEIGHTS AND DIMENSIONS FOR ANALYTIC CALCULATIONS†

Component	Weight (lbs) (kN)
MPC baseplate	3,000 (13.3)
MPC closure lid	10,400 (46.3)
MPC shell	5,900 (26.2)
MPC basket supports and fuel spacers	3,700 (16.5)
Fuel basket	13,000 (57.8)
Fuel	54,000 (240.2)
Total MPC package	90,000 (400.3)
Overpack bottom plate	10,000 (44.5)
Overpack closure plate	8,000 (35.6)
Overpack shell	137,000 (609.4)
Total overpack bounding weight	158,000 (702.8)
Total HI-STAR 100 bounding lift weight	250,000 (1,112)
Item	**Dimension (inch) (mm)**
Overpack Outer Diameter	96 (2,438)
Overpack Length	203.125†† (5,159)
MPC Outer Diameter	68.375 (1,737)
MPC Length	190.5 (4,839)
Overpack Inner Diameter	68.75 (1,746)

† Analytic calculations may use the weights and dimensions in Table 3.2.4 or actual weights and dimensions for conservatism in calculation of safety factors. Finite element analyses use other bounding weights or weights calculated based on input weight densities.

†† Overpack length is measured from the bottom surface of the bottom plate to the top surface of the closure plate. The maximum overall length of the overpack is 203.25 inches (5,163 mm), as measured from the bottom of the bottom plate to the top of the top flange. The difference in length has a negligible effect on the calculations.

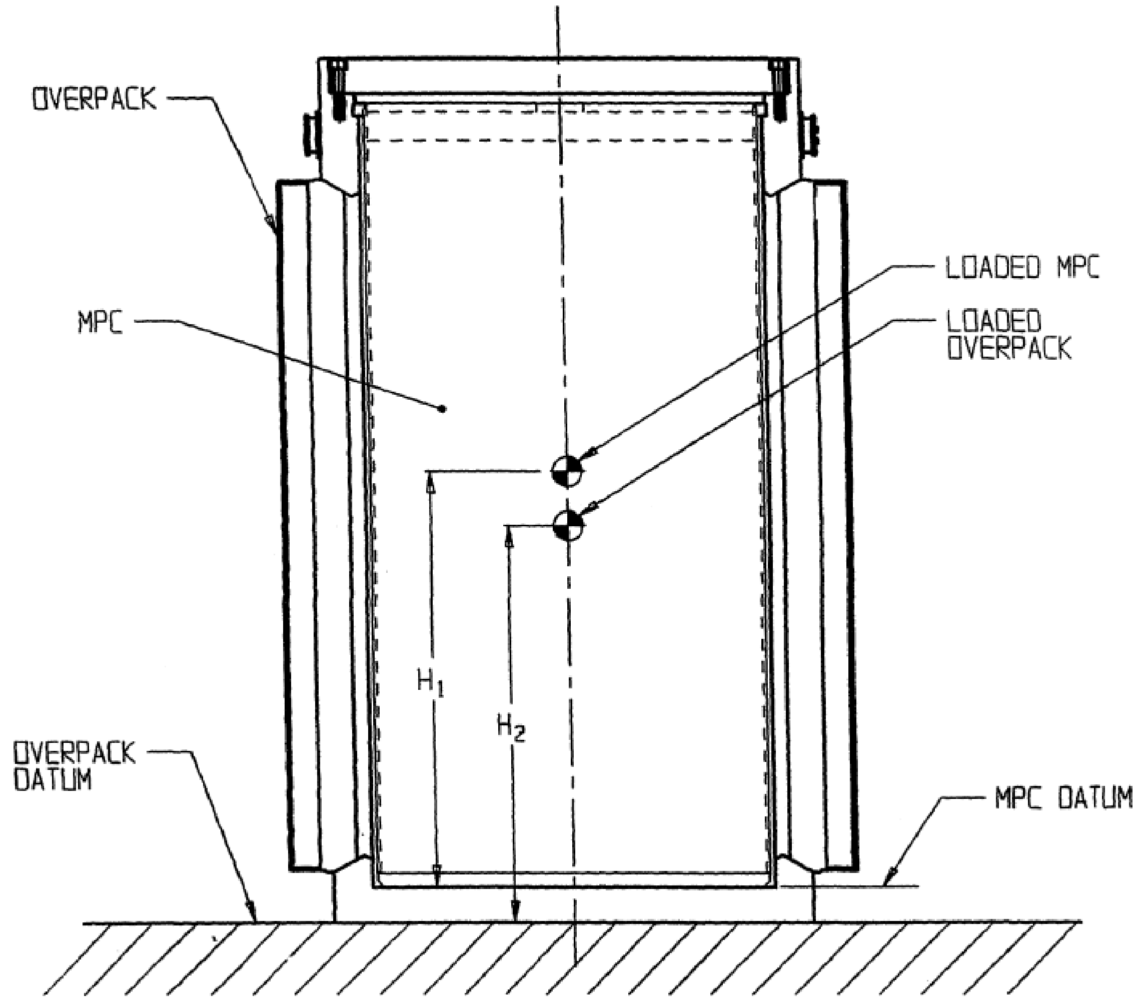

FIGURE 3.2.1; HI-STAR 100 DATUM DEFINITION FOR TABLE 3.2.2

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3.3 MECHANICAL PROPERTIES OF MATERIALS

This section provides the mechanical properties used in the structural evaluation. The properties include yield stress, ultimate stress, modulus of elasticity, Poisson's ratio, weight density, and coefficient of thermal expansion. Values are presented for a range of temperatures; the limits of which are below the off-normal environmental temperature and above the off-normal design temperature.

The materials selected for use in the HI-STAR 100 MPC and overpack are presented on the Bills-of-Material. In this chapter, the materials are divided into two categories, structural and nonstructural. Structural materials are materials that act as load bearing members in the analysis. Materials that do not support mechanical loads are considered nonstructural. For example, while the overpack inner shell is a structural material, Holtite-A (neutron shield) is a nonstructural material.

3.3.1 Structural Materials

3.3.1.1 Alloy X

A hypothetical material termed Alloy X is defined for all MPC structural components. The material properties of Alloy X are the least favorable values from the set of candidate stainless alloys. The purpose of a least favorable material definition is to ensure that all structural analyses are conservative, regardless of the actual MPC material. For example, when evaluating the stresses in the MPC, it is conservative to work with the minimum values for yield strength and ultimate strength. This guarantees that the material used for fabrication of the MPC is of equal or greater strength than the hypothetical material used in the analysis. In the structural evaluation, the only property for which it is not always conservative to use the set of minimum values is the coefficient of thermal expansion. Two sets of values for the coefficient of thermal expansion are specified, a minimum set and a maximum set. For each analysis, the set of coefficients, minimum or maximum, that causes the most adverse result for the cask system is used. Table 3.3.1 lists the numerical values for the material properties of Alloy X versus temperature. These values, taken from the ASME Code, Section II, Part D [3.3.1], are used to complete all structural analyses. The maximum temperatures in some MPC components may exceed the allowable limits of temperature during short time duration loading operations, off-normal transfer operations, or storage accident events. However, under no scenario does the maximum temperature of Alloy X material used in the confinement boundary exceed 1000ºF (538ºC). As shown in ASME Code Case N-47-33 (Class 1 Components in Elevated Temperature Service, 1995 Code Cases, Nuclear Components), the strength properties of austenitic stainless steels do not change due to exposure to 1000ºF (538ºC) temperature for up to 10,000 hours. Therefore, there is no significant effect on mechanical properties of the confinement or basket material during the short time duration loading. A further description of Alloy X, including the materials from which it is derived, is provided in Appendix 1.A.

Two properties of Alloy X which are not included in Table 3.3.1 are weight density and Poisson's ratio. These properties are assumed constant for all structural analyses because there is no significant variation with temperature. The values used are shown in the table below.

PROPERTY	VALUE
Weight Density (lb/in^3) (g/cm^3)	0.290 (8.03)
Poisson's Ratio	0.30

3.3.1.2 Carbon Steel, Low-Alloy and Nickel Alloy Steel

The carbon steels in the HI-STAR 100 System are SA516 Grade 70 and SA515 Grade 70. The nickel alloy and low alloy steels are SA203-E and SA350-LF3, respectively. These steels are not constituents of Alloy X. The material properties of SA516 Grade 70 and SA515 Grade 70 are presented in Table 3.3.2. The material properties of SA203-E and SA350-LF3 are given in Table 3.3.3.

Two properties of these steels which are not included in Tables 3.3.2 and 3.3.3 are weight density and Poisson's ratio. These properties are assumed constant for all structural analyses because there is no significant variation with temperature. The values used are shown in the table below.

PROPERTY	VALUE
Weight Density (lb/in^3) (g/cm^3)	0.283 (7.83)
Poisson's Ratio	0.30

3.3.1.3 Bolting Materials

Material properties of the bolting materials used in the HI-STAR 100 System are given in Table 3.3.4.

3.3.1.4 Weld Material

All weld materials utilized in the welding of the Code components will comply with the provisions of the appropriate ASME subsection (e.g., Subsection NB for the enclosure vessel) and Section IX. All non-code welds shall also be made using weld procedures which meet Section IX of the ASME Code, as practicable. The minimum tensile strength of the weld wire and filler material (where applicable) will be equal to or greater than the tensile strength of the base metal listed in the ASME Code.

3.3.2 Nonstructural Materials

3.3.2.1 Neutron Shield

The neutron shield in the overpack is not considered as a structural member of the HI-STAR 100 System. Its load carrying capacity is neglected in all structural analyses except where such omission would be nonconservative. The only material property of the neutron shield which is important to the structural evaluation is its weight density (1.63g/cm^3).

3.3.2.2 Neutron Absorber

The fuel basket neutron absorber is not a structural member of the HI-STAR 100 System. Its load carrying capacity is neglected in all structural analyses. The only material property of neutron absorber which is important to the structural evaluation is its weight density. As the MPC fuel baskets can be constructed with neutron absorber panels of variable areal density, the weight that produces the most severe cask load is assumed in each analysis (density 2.644 g/cm^3).

3.3.2.3 Aluminum Conduction Inserts

Aluminum conduction inserts are located between the fuel basket and MPC vessel. They are thin, flexible elements whose sole function is to transmit heat. They are not credited with any structural load capacity, and are shaped to provide negligible resistance to basket thermal expansion. The total weight of the aluminum inserts is less than 1,000 lb (4.45 kN) per MPC.

Table 3.3.1
ALLOY X MATERIAL PROPERTIES

Temp. (°F) (°C)	Alloy X				
	S_y[1]	S_u[†]	α_{min}	α_{max}	E
-40 (-40)	30.0 [33.0] (206.8 [227.5])	75.0 *[70.0]* (517.1 [482.6])	8.54 (15.37)	8.55 (15.39)	28.82 (198.7)
100 (38)	30.0 [33.0] (206.8 [227.5])	75.0 *[70.0]* (517.1 [482.6])	8.54 (15.37)	8.55 (15.39)	28.14 (194.0)
150 (66)	27.5 [30.25] (189.6 [208.6])	73.0 *[68.1]* (503.3 [469.5])	8.64 (15.55)	8.67 (15.61)	27.87 (192.2)
200 (93)	25.0 [27.5] (172.4 [189.6])	71.0 *[66.2]* (489.5 [456.4])	8.76 (15.77)	8.79 (15.82)	27.6 (190.3)
250 (121)	23.75 [26.12] (163.8 [180.1])	68.5 *[63.85]* (472.3 [440.23])	8.88 (15.98)	8.9 (16.0)	27.3 (188.2)
300 (149)	22.5 [24.75] (155.1 [170.6])	66.0 *[61.5]* (455.0 [424.0])	8.97 (16.15)	9.0 (16.2)	27.0 (186.2)
350 (177)	21.6 [23.76] (148.9 [163.8])	65.2 *[60.75]* (449.5 [418.86])	9.10 (16.38)	9.11 (16.40)	26.75 (184.4)
400 (204)	20.7 [22.77] (142.7 [157.0])	64.4 *[60.0]* (444.0 [413.7])	9.19 (16.54)	9.21 (16.58)	26.5 (182.7)
450 (232)	20.05 [22.06] (138.2 [152.1])	64.0 *[59.65]* (441.3 [411.27])	9.28 (16.70)	9.32 (16.78)	26.15 (180.3)
500 (260)	19.4 [21.34] (133.8 [147.1])	63.5 *[59.3]* (437.8 [408.8])	9.37 (16.87)	9.42 (16.96)	25.8 (177.9)
550 (288)	18.8 [20.68] (129.6 [142.6])	63.3 *[59.1]* (436.4 [407.5])	9.45 (17.01)	9.50 (17.10)	25.55 (176.2)
600 (316)	18.2 [20.02] (125.5 [138.0])	63.1 *[58.9]* (435.0 [406.1])	9.53 (17.15)	9.6 (17.28)	25.3 (174.4)
650 (343)	17.8 [19.58] (122.7 [135.0])	62.8 *[58.6]* (433.0 [404.0])	9.61 (17.3)	9.69 (17.44)	25.05 (172.7)

[1] Values in the brackets correspond to yield stress of MPC Lids which are 10% greater than the minimum yield stress values tabulated in Table 1.A.3. These higher values are only credited for the stress analysis of the MPC Lid lifting holes in Subsection 3.4.3.3.

[†] The ultimate stress of Alloy X is dependent on the product form of the material (i.e., forging vs. plate). Values in brackets are based on SA-336 forged materials (type F304, F304LN, F316, and F316LN), which are used solely for the one-piece construction MPC lids. All other values correspond to SA-240 plate material.

700 (371)	17.3 [19.03] (119.3 (131.21))	62.5 *[58.4]* (430.9 [402.6])	9.69 (17.44)	9.76 (17.57)	24.8 (171.0)
750 (399)	16.9 [18.59] (116.5 [128.17])	62.2 *[58.1]* (428.8 [400.6])	9.76 (17.57)	9.81 (17.66)	24.45 (168.6)
800 (427)	16.6 [18.26] (114.4 [125.90])	61.7 *[57.6]* (425.4 [397.1])	9.82 (17.68)	9.90 (17.82)	24.1 (166.2)

Definitions:

S_y = Yield Stress (ksi) (MPa)
α = Mean Coefficient of thermal expansion (in./in. per degree F x 10^{-6}) (mm/mm per Kelvin x 10^{-6})
S_u = Ultimate Stress (ksi) (MPa)
E = Young's Modulus (psi x 10^6) (GPa)

Notes:

1. Source for S_y values is Table Y-1 of [3.3.1].
2. Source for S_u values is Table U of [3.3.1].
3. Source for α_{min} and α_{max} values is Table TE-1 of [3.3.1].
4. Source for E values is material group G in Table TM-1 of [3.3.1].

Table 3.3.2
SA516 AND SA515, GRADE 70 MATERIAL PROPERTIES

Temp. (ºF) (ºC)	SA516 and SA515, Grade 70			
	S_y	S_u	α	E
-40 (-40)	38.0 (262.0)	70.0 (482.6)	5.53 (9.95)	29.95 (206.50)
100 (38)	38.0 (262.0)	70.0 (482.6)	5.53 (9.95)	29.34 (202.29)
150 (66)	36.3 (250.3)	70.0 (482.6)	5.71 (10.28)	29.1 (200.6)
200 (93)	34.6 (238.6)	70.0 (482.6)	5.89 (10.60)	28.8 (198.6)
250 (121)	34.15 (235.46)	70.0 (482.6)	6.09 (10.96)	28.6 (197.2)
300 (149)	33.7 (232.4)	70.0 (482.6)	6.26 (11.27)	28.3 (195.1)
350 (177)	33.15 (228.56)	70.0 (482.6)	6.43 (11.57)	28.0 (193.0)
400 (204)	32.6 (224.8)	70.0 (482.6)	6.61 (11.90)	27.7 (191.0)
450 (232)	31.65 (218.22)	70.0 (482.6)	6.77 (12.19)	27.5 (189.6)
500 (260)	30.7 (211.7)	70.0 (482.6)	6.91 (12.44)	27.3 (188.2)
550 (288)	29.4 (202.7)	70.0 (482.6)	7.06 (12.71)	27.0 (186.2)
600 (316)	28.1 (193.7)	70.0 (482.6)	7.17 (12.91)	26.7 (184.1)
650 (343)	27.6 (190.3)	70.0 (482.6)	7.30 (13.14)	26.1 (180.0)
700 (371)	27.4 (188.9)	70.0 (482.6)	7.41 (13.34)	25.5 (175.8)
750 (399)	26.5 (182.7)	69.3 (477.8)	7.50 (13.50)	24.85 (171.33)

Definitions:

S_y = Yield Stress (ksi) (MPa)
α = Mean Coefficient of thermal expansion (in./in. per degree F x 10^{-6}) (mm/mm per Kelvin x 10^{-6})
S_u = Ultimate Stress (ksi) (MPa)
E = Young's Modulus (psi x 10^6) (GPa)

Notes:
1. Source for S_y values is Table Y-1 of [3.3.1].
2. Source for S_u values is Table U of [3.3.1].
3. Source for α values is material group C in Table TE-1 of [3.3.1].
4. Source for E values is "Carbon steels with C $\leq$ 0.30%" in Table TM-1 of [3.3.1].

Table 3.3.3

SA350-LF3 AND SA203-E MATERIAL PROPERTIES

Temp. (ºF) (ºC)	SA350-LF3			SA350-LF3/SA203-E		SA203-E		
	S_m	S_y	S_u	E	α	S_m	S_y	S_u
-20 (-29)	23.3 (160.6)	37.5 (258.6)	70.0 (482.6)	28.2 (194.4)	---	23.3 (160.6)	40.0 (275.8)	70.0 (482.6)
100 (38)	23.3 (160.6)	37.5 (258.6)	70.0 (482.6)	27.6 (190.3)	6.27 (11.29)	23.3 (160.6)	40.0 (275.8)	70.0 (482.6)
200 (93)	22.8 (157.2)	34.2 (235.8)	68.5 (472.3)	27.1 (186.8)	6.54 (11.77)	23.3 (160.6)	36.5 (251.6)	70.0 (482.6)
300 (149)	22.2 (153.1)	33.2 (228.9)	66.7 (459.9)	26.7 (184.1)	6.78 (12.20)	23.3 (160.6)	35.4 (244.1)	70.0 (482.6)
400 (204)	21.5 (148.2)	32.2 (222.0)	64.6 (445.4)	26.1 (180.0)	6.98 (12.56)	22.9 (157.9)	34.3 (236.5)	68.8 (474.4)
500 (260)	20.2 (139.3)	30.3 (208.9)	60.7 (418.5)	25.7 (177.2)	7.16 (12.89)	21.6 (148.9)	32.4 (223.4)	64.9 (447.5)
600 (316)	18.5 (127.6)	-	-	-	-	-	-	-
700 (371)	16.8 (115.8)	-	-	-	-	-	-	-

Definitions:

S_m = Design Stress Intensity (ksi) (MPa)
S_y = Yield Stress (ksi) (MPa)
S_u = Ultimate Stress (ksi) (MPa)
α = Coefficient of Thermal Expansion (in./in. per degree F x 10^{-6}) (mm/mm per Kelvin x 10^{-6})
E = Young's Modulus (psi x 10^6) (GPa)

Notes:

1. Source for S_m values is ASME Code, Table 2A of [3.3.1].
2. Source for S_y values is ASME Code, Table Y-1 of [3.3.1].
3. Source for S_u values is ratioing S_m values.
4. Source for α values is material group E in Table TE-1 of [3.3.1].
5. Source for E values is material group B in Table TM-1 of [3.3.1].

Table 3.3.4

SB637-N07718, SA564-630, AND SA705-630 MATERIAL PROPERTIES

Temp. (ºF) (ºC)	SB637-N07718				
	S_y	S_u	E	α	S_m
-100 (-73)	150.0 (1034.2)	185.0 (1275.5)	29.9 (206.2)	---	50.0 (344.7)
-20 (-29)	150.0 (1034.2)	185.0 (1275.5)	---	---	50.0 (344.7)
70 (21)	150.0 (1034.2)	185.0 (1275.5)	29.0 (206.2)	7.05 (12.69)	50.0 (344.7)
100 (38)	150.0 (1034.2)	185.0 (1275.5)	---	7.08 (12.74)	50.0 (344.7)
200 (93)	144.0 (992.8)	177.6 (1224.5)	28.3 (195.1)	7.22 (13.00)	48.0 (330.9)
300 (149)	140.7 (970.1)	173.5 (1196.2)	27.8 (191.7)	7.33 (13.19)	46.9 (323.4)
400 (204)	138.3 (953.5)	170.6 (1176.2)	27.6 (190.3)	7.45 (13.41)	46.1 (317.8)
500 (260)	136.8 (943.2)	168.7 (1163.1)	27.1 (186.8)	7.57 (13.63)	45.6 (314.4)
600 (316)	135.3 (932.9)	166.9 (1150.7)	26.8 (184.8)	7.67 (13.81)	45.1 (311.0)
SA705-630/SA564-630 (Age Hardened at 1075ºF (579ºC))					
Temp. (ºF) (ºC)	S_y	S_u	E	α	-
200 (93)	115.6 (797.0)	145.0 (999.7)	28.5 (196.5)	5.9 (10.62)	-
300 (149)	110.7 (763.2)	145.0 (999.7)	27.9 (192.4)	5.9 (10.62)	-
SA705-630/SA564-630 (Age Hardened at 1150ºF (621ºC))					
200 (93)	97.1 (669.5)	135.0 (930.8)	28.5 (196.5)	5.9 (10.62)	-
300 (149)	93.0 (641.2)	135.0 (930.8)	27.9 (192.4)	5.9 (10.62)	-

Definitions:

S_m = Design stress intensity (ksi) (MPa)
S_y = Yield Stress (ksi) (MPa)
α = Mean Coefficient of thermal expansion (in./in. per degree F x 10^{-6}) (mm/mm per Kelvin x 10^{-6})
S_u = Ultimate Stress (ksi) (MPa)
E = Young's Modulus (psi x 10^6) (GPa)

Notes:

1. Source for S_m values is Table 4 of [3.3.1].

2. Source for S_y values is ratioing design stress intensity values.
3. Source for S_u values is ratioing design stress intensity values.
4. Source for α values is Tables TE-1 and TE-4 of [3.3.1], as applicable.
5. Source for E values is Table TM-1 of [3.3.1].

TABLE 3.3.5
SA36 AND CARBON STEEL MATERIAL PROPERTIES

Temp. (Deg. F) (Deg. C)	SA36 AND CARBON STEEL			
	S_y	S_u	α	E
-40 (-40)	36.0 (248.2)	58.0 (399.9)	---	29.95 (206.50)
100 (38)	36.0 (248.2)	58.0 (399.9)	5.53 (9.95)	29.34 (202.29)
150 (66)	34.4 (237.2)	55.4 (382.0)	5.71 (10.28)	29.1 (200.6)
200 (93)	32.8 (226.1)	52.8 (364.0)	5.89 (10.60)	28.8 (198.6)
250 (121)	32.35 (223.0)	52.1 (359.2)	6.09 (10.96)	28.6 (197.2)
300 (149)	31.9 (219.9)	51.4 (354.4)	6.26 (11.27)	28.3 (195.1)
350 (177)	31.35 (216.2)	50.5 (348.2)	6.43 (11.57)	28.0 (193.0)
400 (204)	30.8 (212.4)	49.6 (342.0)	6.61 (11.90)	27.7 (191.0)
450 (232)	29.95 (206.5)	48.3 (333.0)	6.77 (12.19)	27.5 (189.6)
500 (260)	29.1 (200.6)	46.9 (323.4)	6.91 (12.44)	27.3 (188.2)
550 (288)	27.85 (192.0)	44.9 (309.6)	7.06 (12.71)	27.0 (186.2)
600 (316)	26.6 (183.4)	42.9 (295.8)	7.17 (12.91)	26.7 (184.1)
650 (343)	26.1 (180.0)	42.1 (290.3)	7.30 (13.14)	26.1 (180.0)
700 (371)	25.9 (178.6)	41.7 (287.5)	7.41 (13.34)	25.5 (175.8)

Definitions:

S_y = Yield Stress (ksi) (MPa)
α = Mean Coefficient of thermal expansion (in./in. per degree F x 10^{-6}) (mm/mm per Kelvin x 10^{-6})
S_u = Ultimate Stress (ksi) (MPa)
E = Young's Modulus (psi x 10^6) (GPa)

Notes:

1. Source for S_y values is Table Y-1 of [3.3.1].
2. Source for S_u values is ratioing S_y values.
3. Source for α values is material group C in Table TE-1 of [3.3.1].
4. Source for E values is "Carbon steels with C less than or equal to 0.30%" in Table TM-1 of [3.3.1].

3.4 GENERAL STANDARDS FOR CASKS

3.4.1 Chemical and Galvanic Reactions

In this subsection, it is shown that there is no credible mechanism for significant chemical or galvanic reactions in the HI-STAR 100 System during long-term storage operations.

The MPC, which is filled with helium, provides a nonaqueous and inert environment. Insofar as corrosion is a long-term time-dependent phenomenon, the inert gas environment in the MPC precludes the incidence of corrosion during storage on the ISFSI. Furthermore, the only dissimilar material groups in the MPC are: (1) neutron absorber material and stainless steel and (2) aluminum and stainless steel. Neutron absorber materials and stainless steel have been used in close proximity in wet storage for over 30 years. Many spent fuel pools at nuclear plants contain fuel racks, which are fabricated from neutron absorber materials and stainless steel materials, with geometries similar to the HI-STAR 100 MPC. Not one case of chemical or galvanic degradation has been found in fuel racks built by Holtec. This experience provides a sound basis to conclude that corrosion will not occur in these materials. Additionally, the aluminum conduction inserts and stainless steel basket are very close on the galvanic series chart. Aluminum, like other metals of its genre (e.g., titanium and magnesium) rapidly passivates in an aqueous environment, leading to a thin ceramic (Al_2O_3) barrier which renders the material essentially inert and corrosion-free over long periods of application. The physical properties of the material, e.g., thermal expansion coefficient, diffusivity, and thermal conductivity, are essentially unaltered by the exposure of the aluminum metal stock to an aqueous environment.

In order to minimize the incidence of aluminum water reaction inside the MPC during fuel loading operation (when the MPC is flooded with pool water) all aluminum surfaces are pre-passivated or anodized before installation of neutron absorber material or conduction inserts in the MPC. The aluminum in the optional heat conduction elements will quickly passivate in air and in water to form a protective oxide layer that prevents any significant hydrogen production during MPC cask loading and unloading operations. The aluminum in the neutron absorber material, particularly in the core area, will also react with water to generate hydrogen gas. The exact rate of generation and total amount of hydrogen generated is a function of a number of variables (see Section 1.2.1.3.1) and cannot be predicted with any certainty. Therefore, to preclude the potential for hydrogen ignition during lid welding or cutting, the operating procedures in Chapter 8 require monitoring for combustible gas and either exhausting or purging the space beneath the MPC lid with an inert gas during these activities. Once the MPC cavity is drained, dried, and backfilled with helium, the source of hydrogen gas (the aluminum-water reaction) is eliminated.

The HI-STAR 100 overpack combines low alloy and nickel alloy steels, carbon steels, neutron and gamma shielding materials, thermal expansion foam, and bolting materials. All of these materials have a long history of nongalvanic behavior within close proximity of each other. The internal and external steel surfaces of each of the overpacks are sandblasted and coated to preclude surface oxidation. Therefore, chemical or galvanic reactions involving the overpack materials are highly unlikely and are not expected.

In accordance with NRC Bulletin 96-04 [3.4.7], a review of the potential for chemical, galvanic, or other reactions among the materials of the HI-STAR 100 System, its contents and the operating environments which may produce adverse reactions has been performed. Table 3.4.2 provides a listing of the materials of fabrication for the HI-STAR 100 System and evaluates the performance of the material in the expected operating environments during short-term loading/unloading operations and long-term storage operations. As a result of this review, no operations were identified which could produce adverse reactions beyond those conditions already analyzed in this FSAR.

3.4.2 Positive Closure

There are no quick-connect/disconnect ports in the confinement boundary of the HI-STAR 100 System. The all welded design of the MPC enclosure vessel precludes access to the stored nuclear fuel without use of special equipment for disconnecting Alloy X pressure vessel parts. The only access to the MPC is through the closure plate, which weighs over 7,000 pounds (31 kN). The closure plate is fastened to the overpack with an array of large bolts. Inadvertent opening of the overpack is not feasible; opening an overpack requires mobilization of special tools and heavy-load lifting equipment.

3.4.3 Lifting Devices

As required by Reg. Guide 3.61, in this subsection, analyses for all lifting operations applicable to the deployment of a HI-STAR 100 System are presented to demonstrate compliance with applicable codes and standards.

The HI-STAR 100 System has the following types of lifting devices: lifting trunnions located on the overpack top flange; threaded holes for eyebolts to lift the overpack closure plate; lifting lugs for the MPC enclosure vessel; and threaded holes for eyebolts for lifting a loaded MPC or the MPC top lid.

The evaluation of the adequacy of the lifting devices entails careful consideration of the applied loading and associated stress limits. The load combination D+H, where H is the "handling load", is the generic case for all lifting adequacy assessments. The term D denotes the dead load. Quite obviously, D must be taken as the bounding value of the dead load of the component being lifted. Table 3.2.4 gives bounding weights. In all lifting analyses considered in this document, the handling load H is assumed to be 0.15D. In other words, the inertia amplifier during the lifting operation is assumed to be equal to 0.15g. This value is consistent with the guidelines of the Crane Manufacturer's Association of America (CMAA), Specification No. 70, 1988, Section 3.3, which stipulates a dynamic factor equal to 0.15 for slowly executed lifts. Thus, the "apparent dead load" of the component for stress analysis purposes is $D^* = 1.15D$. Unless otherwise stated, all lifting analyses in this report use the "apparent dead load", D^*, in the lifting analysis.

Analysis methodology to evaluate the adequacy of the lifting device may be analytical or numerical. For the analysis of the trunnion, an accepted conservative technique for computing the bending stress is to assume that the lifting force is applied at the tip of the trunnion “cantilever” and that the stress state is fully developed at the base of the cantilever. This conservative technique, recommended in NUREG-1536, is applied to all trunnion analyses presented in this FSAR.

In general, the stress analysis to establish safety pursuant to NUREG-0612, Regulatory Guide 3.61, and the ASME Code requires evaluation of three discrete zones which may be referred to as (i) the trunnion, (ii) the trunnion/component interface, hereinafter referred to as Region A, and (iii) the rest of the component, specifically the stressed metal zone adjacent to Region A, herein referred to as Region B.

Stress limits germane to each of the above three areas are discussed below:

i. Trunnion: NUREG-0612 requires that under the "apparent dead load", D^*, the maximum primary stress in the trunnion be less than 10% of the trunnion material ultimate strength *and* less than 1/6th of the trunnion material yield strength. In otherwords, the maximum moment and shear force developed in the trunnion cantilever is less than 1/6 of the moment and shear force corresponding to incipient plasticity, and less than 1/10 of the flexural collapse moment or ultimate shear force for the section.

ii. Region A: Trunnion/Component Interface: Stresses in Region A must meet ASME Code Level A limits under applied load D*. Additionally, Regulatory Guide 3.61 requires that the maximum primary stress under $3D^*$ be less than the yield strength of the weaker of the two materials at the trunnion/component interface. In cases involving section bending, the developed section moment must be compared against the plastic moment at yield. Typically, the stresses in the component in the vicinity of the trunnion/component interface are higher than elsewhere. However, exceptional situations exist. For example, when lifting a loaded MPC, the MPC baseplate, which supports the entire weight of the fuel and the fuel basket, is a candidate location for high stress even though it is far removed from the lifting location (which is located in the top lid).

iii. Region B: This region constitutes the remainder of the component where the stress limits under the concurrent action of the apparent dead load D* and other mechanical loads that may be present during handling (e.g. internal pressure) are required to meet Level A Service Limits.

In summary, both Region A and Region B are required to meet the stress limits corresponding to ASME Level A under the load D*. Additionally, portions of the component that may experience high stress during the lift are subject to the stress criterion of Regulatory Guide 3.61, which requires satisfaction of yield strength as the limit when the sole applied load is 3D*. In general, all locations of high stress in the component under D* must also be checked for compliance with ASME Code Level A stress limits.

Unless explicitly stated otherwise, all analyses of lifting operations presented in this report follow the load definition and allowable stress provisions of the foregoing. Consistent with the practice adopted throughout this chapter, results are presented in dimensionless form, as safety factors, defined as

$$\text{Safety Factor, } SF = \frac{\text{Allowable Stress in the Region Considered}}{\text{Computed Maximum Stress in the Region}}$$

It should be emphasized that the safety factor, SF, defined in the foregoing, represents the *additional margin* that is over any beyond the margin built into NUREG 0612 (e.g. a factor of 10 on ultimate strength or 6 on yield strength).

In the following subsections, we briefly describe each of the lifting analyses performed to demonstrate compliance with regulations. Summary results are presented for each of the analyses.

It is recognized from the discussion in the foregoing that stresses in Region A are subject to two distinct criteria, namely Level A stress limits under D* and other loading that may be present (such as pressure) and yield strength at 3D*. We will use the "3D*" identifier whenever the Regulatory Guide 3.61 load case (the stresses must be bounded by the yield point at 3D*) is the applied loading.

All of the lifting analyses for the overpack reported in this subsection are designated as Load Case 03 in Table 3.1.5. All of the lifting analyses for the MPC reported in this subsection are designated as Load Case E2 in Table 3.1.4. In Subsection 3.4.4, a finite element analysis of the entire overpack is undertaken and results for Load Case 03 (Vertical Handling) in Table 3.1.5 obtained. The results for safety factors from the general finite element model are presented in a later subsection.

3.4.3.1 Overpack Lifting Trunnion Analysis

The lifting trunnion for the HI-STAR 100 overpack is presented in Holtec Drawing 3913 (Section 1.5 herein).

The two lifting trunnions for HI-STAR 100 are circumferentially spaced at 180 degrees. The trunnions are designed for a two-point lift and are sized to satisfy the aforementioned NUREG-0612 criteria. Figure 3.4.43 shows the overall lifting configuration. Supplement 20 of [3.4.13] contains details of the lifting trunnion stress analysis. It is demonstrated in Supplement 20 of [3.4.13] that the stresses in the trunnions, computed in the manner of the foregoing, comply with NUREG-0612 and Regulatory Guide 3.61 provisions.

Specifically, the following results are obtained:

Safety Factors from HI-STAR 100 Lifting Trunnion Stress Analysis†			
Item	Value (ksi) or (lb) or (lb-in) (MPa, kN, kN-mm)	Allowable (ksi) or (lb) or (lb.-in.) (MPa, kN, kN-mm)	Safety Factor
Bending stress (Comparison with Yield Stress/6)	17.3 (119.3)	24.5 (168.9)	1.41
Shear stress (Comparison with Yield Stress/6)	7.4 (51.0)	14.7 (101.4)	1.99
Bending Moment (Comparison with Ultimate Moment/10)	323,000 (36,494)	574,600 (64,921)	1.78
Shear Force (Comparison with Ultimate Force/10)	144,000 (640)	282,000 (1,254)	1.97

† The bounding lifted load is 250,000 lb (1,112 kN) per Table 3.2.4.

We note from the above that all safety factors are greater than 1.0. A factor of safety of exactly 1.0 means that the maximum stress is equal to the yield stress in tension or shear divided by 6, or that the section moment or shear force is equal to the ultimate section moment capacity or section force capacity divided by 10.

3.4.3.2 HI-STAR 100 Overpack Lifting (Load Case 03 in Table 3.1.5)

3.4.3.2.1 Top Flange Under D*

During lifting of a loaded HI-STAR 100, the top flange of the overpack (in which the lift trunnions are located) is identified as a potential location for high stress levels.

Supplement 20 of [3.4.13] contains calculations that analyze the top flange interface with the trunnion under the lifted load D*. The top flange is considered an NB component subject to the lifted load and internal pressure. The membrane stress intensity is computed at the interface and compared to the allowable local membrane stress intensity. The interface region is also conservatively considered as subject to the provisions of NUREG-0612 and the thread shear stress and bearing stress are compared to 1/6 of the top forging yield stress. The following table summarizes the results:

Top Flange – Minimum Safety Factors (Interface with Trunnion)			
Item	Value (ksi) (MPa)	Allowable (ksi) (MPa)	Safety Factor
Bearing Stress (NUREG-0612 Comparison)	3.808 (26.26)	5.975 (41.20)	1.57
Thread Shear Stress (NUREG-0612 Comparison)	3.376 (23.28)	3.585 (24.72)	1.06
Stress Intensity (NB Comparison)	7.857 (54.17)	34.6 (238.6)	4.4

It is noted from the above that all safety factors are greater than 1.0 and that the safety factors for bearing stress and thread shear stress represent the *additional* margin over the factor of safety of 6 on material yielding. A factor of safety of exactly 1.0 means that the maximum stress is equal to the yield stress in tension or shear divided by 6.

3.4.3.2.2 Overpack Top Flange and Baseplate under 3D*

Supplement 26 of [3.4.13] contains finite element analysis and results for the components of the HI-STAR 100 structure that are considered as Region A (namely, the top flange region and baseplate) and evaluated for safety under three times the apparent lifted load (3D*). The overpack baseplate is analyzed using classical plate theory and conservatively assumes that the allowable strengths are determined at the component design temperature rather than at the lower normal operating conditions. The results from Supplement 26 of [3.4.13] for both regions are summarized in the table below.

Overpack Top Flange and Baseplate Minimum Safety Factors (Reg. Guide 3.61 Loading)			
Item	Value (ksi) (MPa)	Allowable (ksi) (MPa)	Safety Factor
Top Flange Membrane Stress Intensity (3D*)	27.44 (189.2)	32.2 (222.0)	1.17
Top Flange Membrane plus Bending Stress Intensity (3D*)	30.0 (206.8)	48.3 (333.0)	1.61
Baseplate Membrane plus Bending Stress Intensity (3D*)	1.452 (10.0)	32.2 (222.0)	22.2

It is noted from the above table that all safety factors of safety are greater than 1.0.

3.4.3.3 MPC Lifting Analysis (Load Case E2 in Table 3.1.4)

The MPC can be inserted or removed from an overpack by lifting bolts that are designed for installation into threaded holes in the top lid. The strength requirements of the bolts and base metal are examined in Supplement 52 of [3.4.14] based on the requirements of NUREG 0612. Sufficiency of thread engagement length and bolt pre-load are also considered in Supplement 52 of [3.4.14]. The MPC top closure is examined in Supplement 14 of [3.4.14], considering the top lid as "Region B", where satisfaction of ASME Code Level A requirements is demonstrated. The same calculation also considers highly stressed regions of the top closure as "Region A" where applied load is 3D*. Supplement 15 of [3.4.14] includes structural analysis of the baseplate under normal handling and subject to the allowable strengths appropriate to a component considered in "Region B". Finally, Supplement 26 of [3.4.13] contains analysis and results for the same baseplate region where the loading is 3D* consistent with the baseplate of the MPC being considered as a "Region A". The definitions of "Region A", "Region B", and "3D*" as they apply to lifting analyses have been introduced at the beginning of this Subsection.

The following table summarizes the results from all of these analyses. As stated earlier, safety factors tabulated in this section represent margins that are over and beyond those implied by the loading magnification mandated in NUREG 0612 or Regulatory Guide 3.61, as appropriate.

Summary of MPC Lifting Analyses-Minimum Safety Factors			
Item	Value of Stress (ksi) (MPa) or Load (lb.) (kN)	Allowable (ksi) (MPa) or Capacity (lb.) (kN)	Safety Factor = Allowable/Value or Capacity/Load
Lifting Bolt Load – NUREG 0612 (Note 1)	103,500 (460)	104,800 (466)	1.013
Top Lid Peripheral Weld Load – (3D*) (Note 2)	310,500 (1,381)	1,055,000 (4,693)	3.40
Top Lid Peripheral Weld Load– "Region B" (Note 2)	460,023 (2,046)	1,055,000 (4,693)	2.29
Baseplate Bending Stress – (3D*) (Note 3)	13.26 (91.42)	20.7 (142.7)	1.56
Baseplate Bending Stress – "Region B" (Note 4)	25.78 (177.7)	28.05 (193.4)	1.09

Notes:

1. Detailed analysis presented in Supplement 52 of [3.4.14]; capacity based on MPC lid having minimum yield strength of 33 ksi at room temperature (see Table 3.3.1).
2. Detailed analysis presented in Supplement 14 of [3.4.14]
3. Detailed analysis presented in Supplement 26 of [3.4.13]
4. Detailed analysis presented in Supplement 15 of [3.4.14]

We note that all factors of safety, except for lid bolt, are greater than 1.0 as required. We also note that the baseplate bending stress calculation in Supplement 15 of [3.4.14] is conservative in that the load from the fuel basket is applied as a uniform pressure over the entire baseplate; in reality, the load is applied as a ring load located near the periphery of the basket. Applying the load in this manner would increase the reported safety factor.

3.4.3.4 Miscellaneous Lifting Analyses

The closure plate of the HI-STAR 100 overpack is lifted using four eyebolt lugs that are threaded into tapped holes in the closure plate. The MPC top lid is lifted using the same tapped holes that are used for lifting a loaded MPC. Figure 8.1.2 identifies the typical lid lifting operation that is indicated as one of the steps in the cask deployment operation.

Supplement 52 of [3.4.14] contains details of the strength qualification of the overpack top closure lifting holes. Qualification is based on the previously discussed NUREG-0612 requirement. Minimum safety factors are summarized in the table below where we note that a safety factor of 1.0 means that the stress is the lessor of yield stress/6 or ultimate stress/10.

Miscellaneous Lid Lifting – Minimum Safety Factors			
Item	Value (lb) (kN)	Capacity (lb) (kN)	Minimum Safety Factor
Overpack Top Closure Lifting Bolt Shear	9,200 (41)	16,880 (75)	1.84
Overpack Top Closure Lifting Bolt Tension	9,200 (41)	20,670 (92)	2.25

Synopses of lifting device, device/component interface, and component stresses, under all contemplated lifting operations for the HI-STAR 100 System have been presented in the foregoing. The results show that all factors of safety are greater than 1.0.

3.4.3.5 Miscellaneous Handling Considerations

Reg. Guide 3.61 and NUREG-1536 do not provide any guidance on the structural requirements for upending or downending operations wherein a location within the body of the cask is used as a pivot or rotational fulcrum. Rotation of the HI-STAR 100 overpack can, however, be carried out using the pocket trunnions (optional) as the pivot axis. Under such a scenario, where each pocket trunnion (optional) is conservatively assumed to support 50% of the loaded HI-STAR 100 weight (125,000 lbs (556 kN)), the pocket is subject to a modest state of stress which can be readily calculated using standard methods.

Because the pocket trunnions (optional) are inserts in the ASME Section III, Subsection NF, Class 3 structure, they do not carry any primary mechanical or inertial loading required of an "NF" part. Therefore, in the storage mode (under 10 CFR 72), the pocket trunnion (optional) can be designated as a non-Code part. The pocket trunnions (optional) do, however, perform an important-to-safety function, namely shielding. By virtue of their structural strength, the pocket trunnions (optional) will maintain their shielding integrity design function under all normal, abnormal, and accident loadings. A detailed discussion of the structural capacity of the pocket trunnions (optional) is provided below.

If the stress limit corresponding to ASME Section III Subsection NF, Level A (normal conditions) is conservatively assumed to be applied, then the factors of safety (as shown in the table below) are quite large. Even larger factors of safety are shown to exist if the structural capacity of the trunnion and welded region are computed using the material ultimate strength diminished by a lower bound value of weld quality factor (0.5 per "NG" of the Code for groove welds).

Factors-of-Safety in the Pocket Trunnion		
Reference for Permissible Stress Limit Values	Location	
	Pocket Trunnion Material	Pocket Trunnion to Cask Body Weld
NF	12.63	8.71
NG	20.48	13.56

It can be seen from the above table that regardless of the ASME Code reference used for the permissible stress level, the relevant factors-of-safety exceed the minimum required value of 1.0 by large margins.

Finally, it can be readily shown that the pocket trunnions (optional) will maintain their shielding function under the most limiting handling accident scenario. Under a design basis drop event, the pocket trunnions (optional) (300 lbs each (1.33 kN each)) may experience a vertical load from their self-inertia of up to:

300 lbs x 60 g = 18,000 lbs   (1.33 kN x 60 g = 80 kN)

This load is reacted by the surrounding weld and by direct bearing on the overpack body. The geometry of the pocket trunnion (optional) is shown on Drawing 3913; the pocket trunnion insert (optional) is completely encapsulated by the surrounding intermediate shells and radial channels and cannot separate from the overpack. The 18,000 pound load (80 kN) on the pocket trunnion (optional) is much less than the 125,000 pound load (556 kN) applied during pocket trunnion (optional) use as a pivot point and, therefore, is bounded by the pivot point analysis above. The encapsulation, together with support from the peripheral welding, ensures that the shielding function of the pocket trunnion (optional) is maintained.

3.4.4 Heat

Subsection 3.4.4, labeled "Heat" in Regulatory Guide 3.61 is required to contain information on all structural (including thermoelastic) analyses performed in the cask to demonstrate positive safety margins, except for lifting operations that are covered in Subsection 3.4.3 in the preceding. Accordingly, this subsection contains all necessary information on the applied loadings, differential thermal expansion considerations, stress analysis models, and results for all normal and off-normal operations, and for natural phenomena/accident events. Assessment of potential malfunction under "Cold" conditions is required to be presented in Subsection 3.4.5.

As instructed by Regulatory Guide 3.61, the thermal evaluation of the HI-STAR 100 System is reported in Chapter 4.

3.4.4.1 Summary of Pressures and Temperatures

Design pressures and design temperatures for all conditions of storage are listed in Tables 2.2.1 and 2.2.3, respectively. Load Cases F1 (Table 3.1.3) and E4 (Table 3.1.4) are defined to study the effect of differential thermal expansion among the constituent components in the HI-STAR 100 System. Figures 3.4.1 and 3.4.2 provide the defining bounding temperature distributions used for the MPC and overpack finite element thermal stress calculations so as to maximize stresses that develop due to such radial gradients. The distribution T is applied conservatively to analyze its effect on the fuel basket, the enclosure vessel, and the overpack.

3.4.4.2 Differential Thermal Expansion

Consistent with the requirements of Reg. Guide 3.61, Load Cases F1 (Table 3.1.3) and E4 (Table 3.1.4) are defined to study the effect of differential thermal expansion among the constituent components in the HI-STAR 100 System. Tables 4.4.9 to 4.4.11 provide the temperatures necessary to perform the differential thermal expansion analyses for the MPC in the HI-STAR 100 System. The material presented in the remainder of this paragraph demonstrates that a physical interference between discrete components of the HI-STAR 100 System (e.g. overpack and enclosure vessel) will not develop due to differential thermal expansion during any operating condition.

3.4.4.2.1 Normal Hot Environment

Closed form calculations are performed to demonstrate that initial gaps between the HI-STAR 100 overpack and the MPC canister, and between the MPC canister and the fuel basket, will not close due to thermal expansion of the system components under normal, off-normal, and accident cases, defined as F1 and E4 in Tables 3.1.3 and 3.1.4, respectively. To assess this in the most conservative manner, the thermal solutions computed in Chapter 4 are surveyed for the following information.

- The radial temperature distribution in each of the fuel baskets at the location of peak center metal temperature.

- The highest and lowest mean temperatures of the canister shell for the hot environment condition.

- The inner and outer surface temperature of the overpack shell (inner shell, intermediate shells, neutron shield, and outer enclosure shell) at the location of highest and lowest surface temperature (which will produce the lowest mean temperature).

Table 4.4.16 presents the resulting temperatures used in the evaluation of the MPC expansion in the HI-STAR 100 overpack.

Using the temperature information in the above-mentioned tables, simplified thermoelastic solutions of equivalent axisymmetric problems are used to obtain conservative estimates of gap closures. The following procedure, which conservatively neglects axial variations in temperature distribution, is utilized.

1. Use the surface temperature information for the fuel basket to define a parabolic distribution in the fuel basket that bounds (from above) the actual temperature distribution. Using this result, generate a conservatively high estimate of the radial and axial growth of the different fuel baskets using classical closed form solutions for thermoelastic deformation in cylindrical bodies.

2. Use the temperatures obtained for the canister to predict an estimate of the radial and axial growth of the canister to check the canister-to-basket gaps.

3. Use the temperatures obtained for the canister to predict an estimate of the radial and axial growth of the canister to check the canister-to-overpack gaps.

4. Use the overpack surface temperatures to construct a logarithmic temperature distribution (characteristic of a thick walled cylinder) at the location used for canister thermal growth calculations; and use this distribution to predict an estimate of overpack radial and axial growth.

5. For given initial clearances, compute the operating clearances.

The calculation procedure outlined above is used to compute the clearance gaps for the HI-STAR 100 overpack with MPC-24, 32, and 68. The results are summarized in the tables given below for normal storage conditions.

THERMOELASTIC DISPLACEMENTS IN THE MPC AND OVERPACK UNDER HOT TEMPERATURE ENVIRONMENT CONDITION				
CANISTER – FUEL BASKET				
	Radial Direction (in) (mm)		Axial Direction (in) (mm)	
Unit	Initial Clearance	Final Gap	Initial Clearance	Final Gap
MPC-24	0.1875 (4.76)	0.140 (3.56)	2.0 (50.8)	1.77 (45.0)
MPC-32	0.1875 (4.76)	0.144 (3.66)	2.0 (50.8)	1.79 (45.5)
MPC-68	0.1875 (4.76)	0.144 (3.66)	2.0 (50.8)	1.79 (45.5)
CANISTER – OVERPACK				
	Radial Direction (in.) (mm)		Axial Direction (in.) (mm)	
Unit	Initial Clearance	Final Gap	Initial Clearance	Final Gap
MPC-24	0.09375 (2.38)	0.068 (1.73)	0.625 (15.9)	0.482 (12.2)
MPC-32	0.09375 (2.38)	0.07 (1.78)	0.625 (15.9)	0.488 (12.4)
MPC-68	0.09375 (2.38)	0.069 (1.75)	0.625 (15.9)	0.482 (12.2)

It can be verified by referring to the design drawings and the foregoing table, that the clearances between the MPC basket and canister structure, as well as that between the MPC shell and storage overpack, are sufficient to preclude a temperature induced interference from differential thermal expansions under normal operating conditions.

3.4.4.2.2 Fire Accident

The growth of the fuel basket during and after the fire accident is also evaluated. It is shown that under the most conservative set of assumptions the fuel basket does not contact either the canister or the MPC lid due to free thermal growth. Therefore, restraint of free end expansion leading to fuel basket distortion will not occur. Hence, ready retrievability of the fuel will be maintained and the fuel will remain in a subcritical configuration. The table below summarizes the results.

THERMOELASTIC DISPLACEMENTS IN THE MPC AND OVERPACK UNDER FIRE ACCIDENT TEMPERATURE ENVIRONMENT				
CANISTER – FUEL BASKET				
	Radial Direction (in.) (mm)		Axial Direction (in.) (mm)	
Unit	Initial Clearance	Final Gap	Initial Clearance	Final Gap
Bounding MPC	0.1875 (4.76)	0.106 (2.69)	2.0 (50.8)	1.604 (40.74)
CANISTER – OVERPACK				
	Radial Direction (in.) (mm)		Axial Direction (in.) (mm)	
Unit	Initial Clearance	Final Gap	Initial Clearance	Final Gap
Bounding MPC	0.09375 (2.38)	0.052 (1.32)	0.625 (15.88)	0.383 (9.73)

Chapter 11 shows that the fire accident has little effect on the MPC temperatures because of the short duration of the fire and the large thermal inertia of the storage overpack. Therefore, structural evaluation of the MPC under the postulated fire event is not required. The external surfaces of the HI-STAR 100 overpack that are directly exposed to the fire event experience maximum rise in temperature. The outer shell and top closure plate are the external surfaces that are in direct contact with heated air from fire. The table below, extracted from data provided in Chapter 11 (Table 11.2.2), provides maximum bulk temperatures attained.

Component Peak Temperatures due to Storage Fire Event	
Component	Maximum Fire Condition Section Temperature (ºF) (ºC)
Overpack Inner Shell	328 (164)
Overpack Top Flange	524 (273)
Overpack Outer Shell (external skin)	854 (457)
Overpack Baseplate	496 (258)
Overpack Closure Plate	484 (251)
Neutron Shield Inner Surface	314 (157)
Neutron Shield Outer Surface	551 (288)
MPC Shell	364 (184)

The following conclusions are readily reached from the above table.

- The maximum temperature of the ferritic steel material in the body of the HI-STAR 100 overpack is well below 50% of the material melting point. (The melting point of carbon and low alloy steels is approximately 2,750ºF (1,510ºC), per Mark's Standard Handbook, Ninth Edition, pp 6-11.)

- The temperature of the neutron shielding material experiences a gradient across the thickness of the shielding. The shielding material adjacent to the hot outer enclosure shell experiences a local temperature of 551 degrees F (288 degrees C). This means that a limited loss of shielding effectiveness may occur.

- Data published by the Oak Ridge National Laboratories indicates that low stresses from the self-weight of the most heated steel members of HI-STAR 100 (the external skin) ensures that no material rupture will occur. According to the Nuclear System Materials Handbook, TID-2666, ORNL, the time-to-rupture for carbon steels at 1,250ºF (677ºC) and 4,821 psi (33.2 MPa) tensile stress is 520 hours. According to the analyses summarized in Chapter 11, the duration of high temperature in the most heated portion of HI-STAR 100 is well under 1 hour, and the temperature never reaches 900ºF (482ºC).

3.4.4.3 Stress Calculations

This subsection presents calculations of the stresses in the different components of the HI-STAR 100 System from the effects of mechanical load case assembled in Section 3.1. Loading cases for the MPC fuel basket, the MPC enclosure vessel, and the HI-STAR 100 storage overpack are listed in Tables 3.1.3 through 3.1.5, respectively. Detailed analyses for the load cases are presented in labeled references that are listed in the load case tables (Tables 3.1.3, 3.1.4, and 3.1.5). An abbreviated description of each of the analyses is presented in the body of the chapter.

In general, as required by Regulatory Guide 3.61, the comparison of the calculated stresses with their corresponding allowables is presented in Subsection 3.4.4.4. However, for clarity in the narrative in this subsection (3.4.4.3), unnumbered summary tables are presented within the text. However, the key stress comparisons are subsequently reproduced in numbered tables associated with Subsection 3.4.4.4 to provide strict compliance with Regulatory Guide 3.61.

The purpose of the analyses is to provide the necessary assurance that there will be no unacceptable risk of criticality, unacceptable release of radioactive material, unacceptable radiation levels, or impairment of ready retrievability.

For all stress evaluations, the allowable stresses and stress intensities for the various HI-STAR 100 System components are based on bounding high metal temperatures to provide additional conservatism (Table 3.1.17 for the MPC basket and shell, for example). Elastic behavior is assumed for all stress analyses. Elastic analysis is based on the assumption of a linear relationship between stress and strain.

In addition to the loading cases germane to stress evaluations mentioned above, three cases pertaining to the stability of HI-STAR 100 are also considered (Table 3.1.1).

The results of various stress calculations on components are reported in this subsection. The calculations are either performed directly as part of the text, or carried out in a separate calculation report that provides details of strength of materials evaluations or finite element numerical analysis. The specific calculations reported in this subsection are:

1. MPC stress and stability calculations
2. HI-STAR 100 overpack stress and stability calculations

The MPC fuel basket and enclosure vessel have been evaluated for the load combinations in Tables 3.1.3 and 3.1.4. The HI-STAR 100 overpack has also been evaluated for certain limiting load conditions that are germane to the storage and operational modes specified for the system in Tables 3.1.1 and 3.1.5.

MPC stress and stability analyses are considered in Subsection 3.4.4.3.1. Within this subsection, the following analyses are performed:

a. Finite element analysis of the MPC fuel basket and enclosure shell under lateral loads.

b. Finite element and analytical analysis of the enclosure vessel as an ASME Code pressure vessel.
c. Elastic stability and yielding analysis of the MPC fuel basket under lateral and axial compression.
d. Analysis of the MPC baseplate under lateral loads.
e. Analysis of the MPC closure lid under lateral load.
f. Analysis of the fuel support spacers under compression load.
g. Elastic stability and yielding of the MPC enclosure shell under axial and lateral loads.

Overpack stress and stability analyses are considered in Subsection 3.4.4.3.2. Within this subsection, the following analyses are performed:

a. Three-dimensional finite element analysis of the overpack subjected to load cases listed in Table 3.1.5.
b. Consideration of fabrication stresses.
c. Structural analysis of closure bolting for normal operation, top closure puncture, and a postulated accident drop condition.
d. Stress analysis of the overpack closure plate under lateral loads.
e. Elastic stability and yielding of the overpack inner shell.
f. Stress analysis of the enclosure shell and enclosure return under internal pressure.

3.4.4.3.1 MPC Stress and Stability Calculations

The structural function of the MPC in the storage mode is stated in Section 3.1. The calculations presented here demonstrate the ability of the MPC to perform its structural function. Analyses are performed for each of the three MPC designs. The purpose of the analyses is to provide the necessary assurance that there will be no unacceptable risk of criticality, unacceptable release of radioactive material, or impairment of ready retrievability. The following subsections describe the model, individual loads, load combinations, and analysis procedures applicable to the MPC.

3.4.4.3.1.1 Analysis of Load Cases E.3.b, E.3.c (Table 3.1.4) and F2, F.3.b, F.3.c (Table 3.1.3)

The load cases considered herein pertain to lateral loading on the MPC components, namely the fuel basket and the enclosure vessel. For this purpose, a finite element model of the MPC is necessary.

- Description of Finite Element Models of the MPCs under Lateral Loading

A finite element model of each MPC is used to assess the effects of the normal, off normal, and accident loads. The models are constructed using ANSYS [3.4.1], and they are identical to the models used in Holtec's HI-STAR 100 10CFR71 submittal under Docket Number 71-9261. The following model description is common to all MPCs.

The MPC structural model is two-dimensional. It represents a one-inch long cross section of the MPC fuel basket and MPC canister.

The MPC model includes the fuel basket, the basket support structures, and the MPC shell. A basket support is defined as any structural member that is welded to the inside surface of the MPC shell. A portion of the overpack inner surface is modeled to provide the correct restraint conditions for the MPC. Figures 3.4.3 through 3.4.11 show the typical MPC models.

The fuel basket support structure shown in the figures and in the drawings in Section 1.5 is a multi-plate structure consisting of solid shims or support members having two separate compressive load supporting members. For conservatism in the finite element model some dual path compression members (i.e., "V" angles) are simulated as single columns. Therefore, the calculated stress intensities in the fuel basket angle supports are conservatively overestimated in some locations.

The ANSYS model is not intended to resolve the detailed stress distributions in weld areas. Individual welds are not included in the finite element model. A separate analysis for basket welds and for the basket support "V" angles is contained in Supplement 18 of [3.4.14].

No credit is taken for any load support offered by the neutron absorber panels, sheathing, and the aluminum heat conduction elements. Therefore, these so-called non-structural members are not represented in the model. The bounding MPC weight used, however, does include the mass contributions of these non-structural components.

The model is built using five ANSYS element types: BEAM3, PLANE82, CONTAC12, CONTAC26, and COMBIN14. The fuel basket and MPC shell are modeled entirely with two-dimensional beam elements (BEAM3). Plate-type basket supports are also modeled with BEAM3 elements. Eight-node plane elements (PLANE82) are used for the solid-type basket supports. The gaps between the fuel basket and the basket supports are represented by two-dimensional point-to-point contact elements (CONTAC12). Contact between the MPC shell and the overpack is modeled using two-dimensional point-to-ground contact elements (CONTAC26) with an appropriate clearance gap.

For each MPC type, three variations of the finite element model were prepared. The basic model includes only the fuel basket and the enclosure shell (Figures 3.4.3 through 3.4.5) and is used only to study the free thermal expansion due to the temperature field developed in the system. The other two models include a representation of the overpack and are used for the two drop cases considered. Two orientations of the deceleration vector are considered. The 0-degree drop model includes the overpack-MPC interface in the basket orientation illustrated in Figure 3.1.2. The 45-degree drop model represents the overpack-MPC interface with the basket oriented in the manner of Figure 3.1.3. The 0-degree and the 45-degree drop models are shown in Figures 3.4.6 through 3.4.11. Table 3.4.1 lists the element types and number of elements for all models for all fuel storage MPC types.

A contact surface is provided in the models used for drop analyses to represent the overpack inner shell. As the MPC makes contact with the overpack, the MPC shell deforms to mate with the inside surface of the inner shell. The nodes that define the elements representing the fuel basket and the MPC shell are located along the centerline of the plate material. As a result, the line of nodes that forms the perimeter of the MPC shell is inset from the real boundary by a distance that is equal to half of the shell thickness. In order to maintain the specified MPC shell/overpack gap dimension, the radius of the overpack inner shell is decreased by an equal amount in the model.

Contact is simulated using two-dimensional point-to-ground elements (CONTAC26). The surface is tangent to the MPC shell at the initial point of impact and extends 135 degrees on both sides. This is sufficient to capture the full extent of contact between the MPC and the overpack.

The three discrete components of the HI-STAR 100 System, namely the fuel basket, the MPC shell, and the storage overpack are engineered with small diametral clearances which are large enough to permit unconstrained thermal expansion of the three components under the rated (maximum) heat duty condition. A small diametral gap under ambient conditions is also necessary to assemble the system without physical interference between the contiguous surfaces of the three components. The required gap to ensure unrestricted thermal expansion between the basket and the MPC shell is less than 0.1 inch (2.54 mm). This gap, too, will decrease under maximum heat load conditions, but will introduce a physical nonlinearity in the structural events involving lateral loading (such as side drop of the system) under ambient conditions. It is evident from the system design drawings that the fuel basket, which is non-radially symmetric, is in proximate contact with the MPC shell at a discrete number of locations along the circumference. At these locations, the MPC shell, backed by the massive overpack weldment, provides a virtually rigid support line to the fuel basket during lateral drop events. Because the fuel basket, the MPC shell, and the overpack are all three-dimensional structural weldments, their inter-body clearances may be somewhat uneven at different azimuthal locations. As the lateral loading is increased, clearances close at the support locations, resulting in the activation of the support from the overpack.

The bending stresses in the basket and the MPC shell at low lateral loading levels which are too small to close the support location clearances are secondary stresses since further increase in the loading will activate the overpack's support action, mitigating further increase in the stress. Therefore, to compute primary stresses in the basket and the MPC shell under lateral drop events, the gaps should be assumed to be closed. However, in the analysis, we have conservatively assumed that an initial gap of 0.1875" (4.76 mm) exists, in the direction of the applied deceleration, at all support locations between the basket and the shell and the radial gap between the shell and the overpack at the support locations is 3/32" (2.38 mm). In the evaluation of safety factors for the MPC-24, 32, and 68, the total stress state produced by the applied loading on this configuration is conservatively compared with primary stress levels, even though the self-limiting stresses should be considered secondary in the strict definition of the Code. To illustrate the conservatism, we have eliminated the secondary stress (that develops to close the clearances) in the comparison with primary stress allowable values and report safety factors for the MPC-24 that are based only on primary stresses necessary to maintain equilibrium with the inertia forces.

- Description of Individual Loads and Boundary Conditions Applied to the MPCs

The method of applying each individual load to the MPC model is described in this subsection. The individual loads are listed in Table 2.2.14. A free-body diagram of the MPC corresponding to each individual load is given in Figures 3.4.12-3.4.15. In the following discussion, references to vertical and horizontal orientations are made. Vertical refers to the direction along the cask axis, and horizontal refers to a radial direction.

Quasi-static structural analysis methods are used. The effects of any dynamic load factors (DLFs) are included in the final evaluation of safety margins. All analyses are carried out using the design basis decelerations in Table 3.1.2.

The MPC models used for side drop evaluations are shown in Figures 3.4.6 through 3.4.11. In each model, the fuel basket and the enclosure vessel are constrained to move only in the direction that is parallel to the acceleration vector. The overpack inner shell, which is defined by three nodes needed to represent the contact surface, is fixed in all degrees of freedom. The fuel basket, enclosure vessel, and overpack inner shell are all connected at one location by linear springs (see Figure 3.4.6, for example).

(a) Accelerations

During a side impact event, the stored fuel is directly supported by the cell walls in the fuel basket. Depending on the orientation of the drop, 0 or 45 degrees (see Figures 3.4.14 and 3.4.15), either one or two walls support the fuel. The effect of deceleration on the fuel basket and canister metal structure is accounted for by amplifying the gravity field in the appropriate direction. In the finite element model this load is effected by applying a uniformly distributed pressure over the full span of the supporting walls. The magnitude of the pressure is determined by the weight of the fuel assembly (Table 2.1.6), the axial length of the fuel basket support structure, the width of the cell wall, and the impact acceleration. It is assumed that the load is evenly distributed along an axial length of basket equal to the fuel basket support structure. For example, the pressure applied to an impacted cell wall during a 0-degree side drop event is calculated as follows:

$$p = \frac{a_n W}{L \lambda}$$

where:

p = pressure

a_n = ratio of the impact acceleration to the gravitational acceleration

W = weight of a stored fuel assembly

L = axial length of the fuel basket support structure

ℓ = width of a cell wall

For the case of a 45-degree side drop the pressure on any cell wall equals p (defined above) divided by the square root of 2. Figure 3.4.12 shows the details of the fuel assembly pressure load on the fuel basket.

(b) Internal Pressure

Design internal pressure in the MPC model is applied by specifying pressure on the inside surface of the enclosure vessel. The magnitude of the internal pressure applied to the model is taken from Table 2.2.1.

For this load condition, the center of the fuel basket is fixed in all degrees of freedom.

(c) Temperature

Temperature distributions are developed in Chapter 4 and applied as nodal temperatures to the finite element model of the MPC enclosure vessel (confinement boundary). Maximum design heat load has been used to develop the temperature distribution used to demonstrate compliance with ASME Code stress intensity levels. A plot of the applied temperature distribution as a function of radius is shown in Figure 3.4.1. Figure 3.4.13 shows the MPC-68 with the typical boundary conditions for all thermal and pressure load cases.

(d) Handling (Lateral Loading)

As discussed in Subsection 3.1.2.1.2, loads arise on the HI-STAR 100 System from normal handling of the cask (e.g., lateral loads while moving the system to the ISFSI). A 2g lateral acceleration, imposed on the fuel basket/enclosure shell finite element model, is assumed to bound lateral handling loads on the MPC under normal handling conditions (Level A Service Conditions).

- Analysis Procedure

The analysis procedure for this set of load cases is as follows:

1. The stress intensity and deformation field due to the combined loads is determined by the finite element solution.

2. The results for each load combination are compared to allowables. The comparison with allowable values is made in Subsection 3.4.4.4.

3.4.4.3.1.2 Analysis of Load Cases E1.a and E1.c (Table 3.1.4)

Load Cases E1.a and E1.c pertain to the performance of the enclosure vessel as a pressure vessel.

Since the MPC shell is a pressure vessel, the classical Lame's calculations should be performed to demonstrate the shell's performance as a pressure vessel. We note that dead load has an insignificant effect on this stress state. We first perform calculations for the shell under internal pressure. Subsequently, we perform a finite element analysis on the entire confinement boundary as a pressure vessel subject to both internal pressure and temperature gradients. Finally, we perform confirmatory hand calculations to gain confidence in the finite element predictions,

- Lame's Solution for the MPC Shell

The stress from internal pressure is found for normal and accident pressures conditions using classical formulas:

Define the following quantities:

P = pressure, r = MPC radius, and t = shell thickness.

Using classical thin shell theory, the circumferential stress, σ_1 = Pr/t, the axial stress σ_2 = Pr/2t, and the radial stress σ_3 = -P are computed for both normal and accident internal pressures. The results are given in the following table:

Classical Shell Theory Results for Normal and Accident Internal Pressures				
Item	σ_1 (psi) (MPa)	σ_2 (psi) (MPa)	σ_3 (psi) (MPa)	σ_1 - σ_3 (psi) (MPa)
P= 100 psi (0.69 MPa)	6,838 (47.1)	3,419 (23.6)	-100 (-0.69)	6,938 (47.8)
P= 200 psi (1.38 MPa)	13,675 (94.3)	6,838 (47.1)	-200 (-1.38)	13,875 (95.7)

Table 3.1.17 provides the allowable membrane strength for Load Case E1 for Alloy X. We see that a safety factor greater than 1.0 exists for the case of normal and accident pressures.

- Finite Element Analysis (Load Case E1.a and E1.c of Table 3.1.4)

Having performed the classical "thin shell under pressure" evaluation, we now proceed to perform a finite element analysis where the interaction between the end closures and the MPC shell is rigorously modeled.

The MPC shell, the top lid, and the baseplate together form the confinement boundary (enclosure vessel) for storage of spent nuclear fuel. In this section, we evaluate the operating condition consisting of dead weight, internal pressure, and thermal effects for the normal heat condition of storage. The top and bottom plates of the MPC enclosure vessel (EV) are modeled using plane axisymmetric elements, while the shell is modeled using the axisymmetric thin shell element. The thickness of the top lid varies in the two MPC types and can be either a single thick lid, or two dual lids welded around their common periphery; the minimum thickness top lid is modeled in the finite element analysis. As applicable, the results for the MPC top lid are modified to account for the fact that in the dual lid configuration, the two lids act independently under mechanical loading. The temperature distributions for all MPC constructions are nearly identical in magnitude and gradient. Temperature differences across the thickness of both the baseplate and the top lid exist during HI-STAR 100's operations. There is also a thermal gradient from the center of the top lid and baseplate out to the shell wall. The metal temperature profile is essentially parabolic from the centerline of the MPC out to the MPC shell. There is also a parabolic temperature profile along the length of the MPC canister. Figure 3.4.44 shows a sketch of the confinement boundary structure with identifiers A-I (also called locating points) where temperature input data is used to represent a continuous temperature distribution for analysis purposes. The overall dimensions of the confinement boundary are also shown in the figure.

Table 4.4.22 provides the desired temperatures for confinement thermal stress analysis. From the tables, we see that the distribution for the MPC-24 provides the largest temperature gradients in the baseplate (from centerline to outer edge) and in the shell (from the joint at the baseplate to the half-height of the cask). It will be shown later that stress intensities are greatest in these components of the confinement vessel. Therefore, detailed stress analyses are performed only for the MPC-24. Because of the intimate contact between the two lid plates when the MPC lid is a two piece unit, there is no significant thermal discontinuity through the thickness; thermal stresses arising in the MPC top lid will be bounding when there is only a single lid. Therefore, for thermal stresses, results from the analysis that considers the lid as a one-piece unit are used and are amplified to reflect the increase in stress in the dual lid configuration.

Figure 3.4.45 shows details of the finite element model of the top lid (considered as a single piece), canister shell, and baseplate. The top lid is modeled with 40 axisymmetric quadrilateral elements; the weld connecting the lid to the shell is modeled by a single element solely to capture the effect of the top lid attachment to the canister offset from the middle surface of the top lid. The MPC canister is modeled by 50 axisymmetric shell elements, with 20 elements concentrated in a short length of shell appropriate to capture the so-called "bending boundary layer" at both the top and bottom ends of the canister. The remaining 10 shell elements model the MPC canister structure away from the shell ends in the region where stress gradients are lower (from the physics of the problem). The baseplate is modeled by 20 axisymmetric quadrilateral elements. Deformation compatibility at the connections is enforced at the top by the single weld element, and deformation and rotation compatibility at the bottom by additional shell elements between nodes 106-107 and 107-108.

The geometry of the model is listed below (terms are defined in Figure 3.4.45):

H_t = 9.5" (241 mm) (the minimum total thickness lid is assumed)

R_L = 0.5 x 67.25" (1,708 mm) (Bill of Materials for Top Lid, Section 1.5)

L_{MPC} = 190.5" (4,839 mm) (Table 3.2.4)

t_s = 0.5" (12.7 mm)

R_S = 0.5 x 68.375" (1,737 mm)

t_{BP} = 2.5” (63.5 mm)

βL = $2\sqrt{R_s t_s} \approx 12"$ (305 mm) (the "bending boundary layer")

Stress analyses are carried out for two cases as follows:

a. internal pressure = 100 psi (0.69 MPa)

b. internal pressure = 100 psi (0.69 MPa), plus applied temperatures for the MPC-24

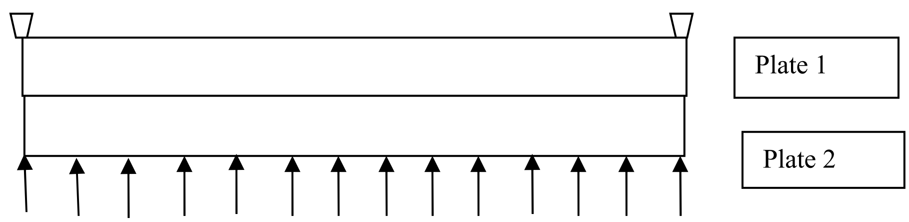
We note that dead weight of the top lid reduces the stresses due to pressure. For example, the equivalent pressure simulating the effect of the weight of the top lid is an external pressure of 3 psi (20.7 kPa), which reduces the pressure difference across the top lid to 97 psi (0.67 MPa). Thus, for conservatism, dead weight of the top lid is neglected to provide additional conservatism in the results. The dead weight of the baseplate, however, adds approximately 0.73 psi (5.0 kPa) to the effective internal pressure acting on the base. The effect of dead weight is still insignificant compared to the 100 psi (0.69 MPa) design pressure, and is therefore neglected. The thermal loading in the confinement vessel is obtained by developing a parabolic temperature profile to the entire length of the MPC canister and to the top lid and baseplate. The temperature data provided at locations A-I in Figures 3.4.44 and 3.4.45 are sufficient to establish the profiles. Through-thickness temperatures are assumed linearly interpolated between top and bottom surfaces of the top lid and baseplate. All material properties and expansion coefficients are considered to be temperature-dependent in the model.

Results for stress intensity are reported for the case of internal pressure alone and for the combined loading of pressure plus temperature (Load Case E1.c in Table 3.1.4). Tables 3.4.7 and 3.4.8 report results at the inside and outside surfaces of the top lid and baseplate at the centerline and at the extreme radius. Canister results are reported in the "bending boundary layer" and at a location near mid-length of the MPC canister. In the tables, the calculated value is the value from the finite element analysis, the categories are P_m = primary membrane; $P_L + P_b$ = local membrane plus primary bending; and $P_L + P_b + Q$ = primary plus secondary stress intensity. The allowable stress intensity value is obtained from the appropriate table in Section 3.1 for Level A conditions, and the safety factor SF is defined as the allowable strength divided by the calculated value. Allowable stresses for Alloy X are taken at 300ºF (149ºC), which bounds the temperatures everywhere except at the mid-length position of the MPC shell (Location I in Figure 3.4.44) during the normal operation. At Location I, the allowable strength is taken at 400ºF (204ºC). The results given in Tables 3.4.7 and 3.4.8 demonstrate the ruggedness of the MPC as a confinement boundary. Since mechanically induced stresses in the top lid are increased when a dual lid configuration is considered, the stress results obtained from an analysis of a single top lid must be corrected to reflect the maximum stress state when a dual lid configuration is considered. The modifications required are based on the following logic:

Consider the case of a simply supported circular plate of thickness h under uniform lateral pressure “q”. Classical strength of materials provides the solution for the maximum stress, which occurs at the center of the plate, in the form:

$\sigma_s = 1.225 q (a/h)^2$ where a is the radius of the plate and h is the plate thickness.

Now consider the MPC simply supported top lid as fabricated from two plates “1” and “2”, of thickness h_1 and h_2, respectively, where the lower surface of plate 2 is subjected to the internal pressure “q”, the upper surface of plate 1 is the outer surface of the helium retention boundary, and the lower surface of plate 1 and the upper surface of plate 2 are in contact. The following sketch shows the dual lid configuration for the purposes of this discussion:



From classical plate theory, if it is assumed that the interface pressure between the two plates is uniform and that both plates deform to the same central deflection, then if

$h_1+h_2 = h$, and if $h_2/h_1 = r$

the following relations exist between the maximum stress in the two individual plates, σ_1, σ_2 and the maximum stress σ_s in the single plate of thickness “h”:

$$\frac{\sigma_1}{\sigma_s} = \frac{(1+r)^2}{(1+r^3)} \qquad \frac{\sigma_2}{\sigma_s} = \frac{(1+r)^2}{(1+r^3)} r$$

Since the two lid thicknesses are the same in the dual lid configuration, r = 1.0 so that the stresses in plates 1 and 2 are both two times larger than the maximum stress computed for the single plate lid having the same total thickness. In Tables 3.4.7 and 3.4.8, bounding results for the dual lid configuration are reported by using these ratios at all locations in the top lid.

- Confirmatory Closed Form Solution

The results in Tables 3.4.7 and 3.4.8 also show that the baseplate and the shell connection to the baseplate are the most highly stressed regions under the action of internal pressure. To confirm the finite element results, we perform an alternate closed form solution using classical plate and shell theory equations that are listed in or developed from the reference Timoshenko and Woinowsky-Krieger, Theory of Plate and Shells, McGraw Hill, Third Edition.

Assuming that the thick baseplate receives little support against rotation from the thin shell, the bending stress at the centerline is evaluated by considering a simply supported plate of radius a and thickness h, subjected to lateral pressure p. The maximum bending stress is given by

$$\sigma = \frac{3(3+\nu)}{8} p \left(\frac{a}{h}\right)^2$$

where:

a = .5 x 68.375” (1,737 mm)

h = 2.5" (63.5 mm)

ν = 0.3 (Poisson’s Ratio)

p = 100 psi (0.69 MPa)

Calculating the stress in the plate gives σ = 23,142 psi (160 MPa).

Now consider the thin MPC shell (t = 0.5" (12.7 mm)) and first assume that the baseplate provides a clamped support to the shell. Under this condition, the bending stress in the thin shell at the connection to the plate is given as

$$\sigma_{Bp} = 3\,p\,\frac{a}{t}\,\frac{(1-\nu/2)}{\sqrt{3}\,(1-\nu^2)^{1/2}} = 10{,}553 \text{ psi } \ (72.76 \text{ MPa})$$

In addition to this stress, there is a component of stress in the shell due to the baseplate rotation that causes the shell to rotate. The joint rotation is essentially driven by the behavior of the baseplate as a simply supported plate; the shell offers little resistance because of the disparity in thickness and will essentially follow the rotation of the thick plate.

Using formulas from thin shell theory, the additional axial bending stress in the shell due to this rotation θ can be written in the form

$$\sigma_{B\theta} = 12\,\beta\,D_s\,\frac{\theta}{t^2}$$

where

$$\theta = pa^3 / 8D\,(1+\nu) * \left(1/_{1+\alpha}\right)$$

and

$$D = \frac{E\,h^3}{12(1-\nu^2)} \qquad E = \text{plate Young's Modulus}$$

$$\alpha = \frac{2\beta a t^3}{h^3(1+\nu)}$$

$$D_s = \frac{E\,t^3}{12\,(1-\nu^2)}$$

$$\beta^2 = \sqrt{3(1-\nu^2)}\,/at$$

Substituting the numerical values gives

$$\sigma_{B\theta} = 40{,}563 \text{ psi } (279.7 \text{ MPa})$$

We note that the approximate solution is independent of the value chosen for Young's Modulus as long as the material properties for the plate and shell are the same.

Combining the two contributions to the shell bending stress gives the total extreme fiber stress in the longitudinal direction as 51,116 psi (352.4 MPa).

The baseplate stress value, 23,142 psi (160 MPa), compares well with the finite element result 20,528 psi (141.5 MPa) (Table 3.4.7). The shell joint stress, 51,116 psi (352.4 MPa), is greater than the finite element result (43,986 psi (303.3 MPa) in Table 3.4.7). This is due to the local effects of the shell-to-baseplate connection offset. That is, the connection between shell and baseplate in the finite element model is at the surface of the baseplate, not at the middle surface of the baseplate. This offset will cause an additional bending moment that will reduce the rotation of the plate and hence, reduce the stress in the shell due to the rotation of the baseplate.

In summary, the approximate closed form solution confirms the accuracy of the finite element analysis in the baseplate region.

3.4.4.3.1.3 Elastic Stability and Yielding of the MPC Basket under Compression Loads (Load Case F3 in Table 3.1.3)

This load case corresponds to the scenario wherein the loaded MPC is postulated to drop causing a compression state in the fuel basket panels.

a. Elastic Stability

Following the provisions of Appendix F of the ASME Code [3.4.3] for stability analysis of Subsection NG structures, (F1331.5(a)(1)), a comprehensive buckling analysis is performed using ANSYS. For this analysis, ANSYS's large deformation capabilities are used. This feature allows ANSYS to account for large nodal rotations in the fuel basket, which are characteristic of column buckling. The interaction between compressive and lateral loading, caused by the deformation, is included in a rigorous manner. The finite element model used for the large deflection analysis of the basket is identical to the model described in Subsection 3.4.4.3.1.1 used for the fuel basket stress analysis. The large deflection option is “turned on” so that equilibrium equations for each load increment are computed based on the current deformed shape. Subsequent to the large deformation analysis, the individual basket panel that is most susceptible to buckling failure is identified by a review of the results. The lateral displacement of a node located at the mid-span of the panel is measured for the range of impact decelerations. The buckling or collapse load is defined as the impact deceleration for which a slight increase in its magnitude results in a disproportionate increase in the lateral displacement. The most critical element is a vertically oriented panel that is subject to a compressive load together with bending moments from adjacent connected lateral basket panels.

The stability requirement for the MPC fuel basket under lateral loading is satisfied if two-thirds of the collapse deceleration load is greater than the design basis horizontal acceleration (Table 3.1.2). Figures 3.4.27 through 3.4.32 are plots of the local lateral displacement versus impact deceleration for the most limiting basket panel. It should be noted that the displacements in Figures 3.4.27 through 3.4.31 are expressed in 1×10^{-1} inch and Figure 3.4.32 is expressed in 1×10^{-2} inch. The plots clearly show that the large deflection collapse load of the MPC fuel basket is greater than 1.5 times the inertia load corresponding to the design basis deceleration for all baskets in all orientations. Thus, the requirements of Appendix F are met for lateral deceleration loading under Subsection NG stress limit s for faulted conditions.

An alternative solution for the stability of the fuel basket panel is obtained using the methodology espoused in NUREG/CR-6322 [3.4.12]. In particular, we consider the fuel basket panels as wide plates in accordance with Section 5 of NUREG/CR-6322. We use eq.(19) in that section with the “K” factor set to the value appropriate to a clamped panel. Material properties are selected corresponding to a metal temperature of 500 degrees F (260 degrees C) which bounds computed metal temperatures at the periphery of the basket. The critical buckling stress is

$$\sigma_{cr} = \left(\pi/K\right)^2 \frac{E}{12\left(1-\nu^2\right)}\left(\frac{h}{a}\right)^2$$

where h is the panel thickness, a is the unsupported panel length, E is the Young's Modulus of Alloy X at 500 degrees F (260 degrees C), ν is Poisson's Ratio, and K=0.65 (per Figure 6 of NUREG/CR-6322).

The MPC-24 has the smallest h/a ratio; the results of the finite element stress analyses under design basis deceleration load show that this basket is subject to the highest compressive load in the panel. Therefore, the critical buckling load is computed using the geometry of the MPC-24. The following table shows the results from the finite element stress analysis and from the stability calculation.

Panel Buckling Results From NUREG/CR-6322			
Item	Finite Element Stress (ksi) (MPa)	Critical Buckling Stress (ksi) (MPa)	Factor of Safety
Stress	13.717 (94.58)	49.22 (339.4)	3.588

For a stainless steel member under an accident condition load, the recommended safety factor is 2.12. We see that the calculated safety factor exceeds this value; therefore, we have independently confirmed the stability predictions of the large deflection analysis based on classical plate stability analysis by employing a simplified method.

Stability of the basket panels, under longitudinal deceleration loading (Load Cases F3.a in Table 3.1.3), is demonstrated in the following manner. From Table 3.2.1 we have the weight of each fuel basket (including sheathing and neutral absorber). The metal areas of the basket bearing on the MPC baseplate can be computed from the drawings in Section 1.5. Dividing weight by area and multiplying by the design basis deceleration from Table 3.1.2 gives the following results.

Fuel Basket Compressive Stress For End Drop (Load Case F3.a)			
Item	Weight (lb) (kN)	Area (sq. inch) (cm^2)	Stress (psi) (MPa)
MPC-68	16,240 (72.2)	250.3 (1,615)	3,893 (26.8)
MPC-24	20,842 (92.7)	357.1 (2,304)	3,502 (24.1)
MPC-32	12,340 (54.9)	195.3 (1,260)	3,791 (26.1)

To demonstrate that elastic instability in the basket panels is not credible, we compute the flat panel buckling stress, σ_{cr}, (critical stress level at which elastic buckling may occur) using the formula in reference [3.4.8].

For elastic stability, Reference [3.4.8] provides the formula for critical axial stress as

$$\sigma_{cr} = \frac{4\,\pi^2\,E}{12\,(1-\nu^2)}\left(\frac{T}{W}\right)^2$$

where T is the panel thickness and W is the width of the panel, E is the Young's Modulus at the metal temperature and ν is the metal Poisson's Ratio. The following table summarizes the calculation for the critical buckling stress using the formula given above:

Elastic Stability Result for a Flat Panel	
Reference Temperature	725 degrees F (385°C)
T (MPC-24)	5/16 inch (7.94 mm)
W	10.777 inch (273.74 mm)
E	24,600,000 psi (169,000 MPa)
Critical Axial Stress	74,781 psi (515.6 MPa)

It is noted the critical axial stress is an order of magnitude greater than the computed basket axial stress reported in the foregoing and demonstrates that elastic stability under longitudinal deceleration load is not a concern.

b. Yielding

The safety factor against yielding of the basket under longitudinal compressive stress from a design basis inertial loading is given by

SF = 17,100 psi/3,893 psi = 4.39 (SF = 117.9 MPa/26.8 MPa = 4.39)

Therefore, plastic deformation of the fuel basket under design basis deceleration is not credible.

3.4.4.3.1.4 MPC Baseplate Analysis (Load Cases E2, E3, E5)

These load cases from Table 3.1.4 consider normal handling, accidental drop, and storage fire.

Minimum safety factors have been reported for Load Case E2 in Subsection 3.4.3 where an evaluation has been performed for stresses under three times the "apparent" load D*. Load Case E3.a provides the limiting accident loading on the baseplate wherein the combined effect of a 60g deceleration plus accident internal pressure is considered. The analysis conservatively neglects support from the overpack during the storage drop accident. During a fire (Load Case E5), the MPC baseplate is subjected to the accident pressure plus dead load, and the fire temperature (which serves only to lower the allowable strengths). All of these analyses are detailed in Supplements 15 and 28 of [3.4.14]; the results are summarized in Table 3.4.12. All reported safety factors are greater than 1.0.

3.4.4.3.1.5 Analysis of the MPC Closure Lid (Load Cases E3, E5)

The closure lid, the closure lid peripheral weld, and the closure ring are examined for maximum stresses developed during the accident drop event and the storage fire.

Analysis of the closure lid for Load Case E1 has been performed previously as part of the finite element analysis of the confinement boundary. Similarly, results for Load Case E2 have been discussed in Subsection 3.4.3 as part of a lifting device. Supplements 14 and 28 of [3.4.14] contain stress analysis of the MPC top closure lid for Load Cases E3 and E5. The closure lid is modeled as a single simply supported plate and is subject to deceleration from an end drop plus appropriate design pressures. Figure 3.E.1 of Supplements 14 and 28 of [3.4.14] shows the configuration considered. Results are presented for both the single and dual lid configuration (in parentheses). For the dual lid configuration, the two plates each support their own amplified weight as simply supported plates under a bottom end drop. The results for minimum safety factor are reported in Table 3.4.13.

3.4.4.3.1.6 Structural Analysis of the Fuel Support Spacers (Load Cases F2 and F3.a)

Upper and lower fuel support spacers are utilized to position the active fuel region of the spent nuclear fuel within the poisoned region of the fuel basket. It is necessary to ensure that the spacers will continue to maintain their structural integrity after an accident event. Ensuring structural integrity implies that the spacer will not buckle under the maximum compressive load, and that the maximum compressive stress will not exceed the compressive strength of the spacer material (Alloy X). Detailed calculations in Supplement 16 of [3.4.14] demonstrate that large structural margins in the fuel spacers are available for the entire range of spacer lengths that may be used in HI-STAR 100 applications (for the various acceptable fuel types). For normal and off-normal operation (Level A Service Condition), a 10g deceleration load is applied (to cover the case of transport wherein the railroad longitudinal design basis g level is 10 (see the HI-STAR 100 SAR, Docket 71-9261)). For accident conditions, a 60g deceleration is the applied loading (Level D Service Condition). The results are summarized in Table 3.4.14.

3.4.4.3.1.7 Enclosure Vessel Stability (Load Case E1.b, E2, E3, and E5 Table 3.1.4)

The MPC shell is examined for elastic/plastic instability due to external pressure or compressive loads introduced as part of these load cases (design external pressure, normal handling, accident vertical drop, and storage fire). Each load component is examined separately. Design external pressure is applied to the outer surface of the EV shell in the MPC model. The magnitude of the external pressure applied to the model is taken from Table 2.2.1. Analysis of the MPC under the external pressure is provided in Supplement 23 of [3.4.13] Analyses are performed using the methodology of ASME Code Case N-284 [3.4.6]. The following stability evaluations are performed in Supplement 23 of [3.4.13] for the MPC shell:

a. A 1.15g compressive handling load.

b. Design basis deceleration inertia load.

c. Accident external pressure plus a 1g compressive dead load.

d. Design external pressure plus a 1g compressive dead load.

The interaction equations for the ASME Code Case N-284 are evaluated and shown to give results less than 1.0 for all of the above conditions. The limiting results from all of the calculations performed are summarized in Table 3.4.15.

The results demonstrate that the MPC shell meets the requirements of Code Case N-284. We note that the stability results presented above are very conservative. The stability analyses in Supplement 23 of [3.4.13] carried out for the MPC shell assumed no axial stiffening from the fuel basket supports that run the full length of the shell. An analysis that included the effect of the stiffening (and therefore, recognized the fact that instability will most likely occur between stiffeners) will give increased safety factors for Load Cases E5 and E1.b.

3.4.4.3.2 Overpack Stress Calculations

The structural functions of the overpack are stated in Section 3.1. The analyses presented here demonstrate the ability of components of the HI-STAR 100 overpack to perform their structural functions in the storage mode. Load cases applicable to the structural evaluation of the HI-STAR 100 overpack are compiled in Table 3.1.5.

The purpose of the analyses is to provide the necessary assurance that the design of the HI-STAR 100 overpack precludes unacceptable release of radioactive material, unacceptable radiation levels, or impairment of ready retrievability of the MPC during system deployment and throughout its service life.

In this subsection, stresses and stress intensities in the HI-STAR 100 overpack due to the combined effects of thermal gradients, pressure, and mechanical loads are presented. The results are obtained from a series of finite element analyses on the complete overpack and separate analyses on overpack components.

3.4.4.3.2.1 Finite Element Analysis – Load Cases 01 to 05 in Table 3.1.5

Load Cases 01 and 05 pertain to demonstration of the overpack helium retention boundary as an ASME "NB" component under Design Pressure and Level A Service Condition thermal loading. Other cases pertain to handling, handling accident, and natural phenomena events. To analyze these load cases, a suitable finite element model of the complete overpack is required.

- Description of Finite Element Model

The purpose of the HI-STAR 100 overpack model is to calculate stresses and stress intensities resulting from the loadings defined in Chapter 2 and compiled into load cases in Table 3.1.5 (including Load Cases 01 and 05).

A three-dimensional finite element model of the HI-STAR 100 overpack is used to assess the effects of normal, off normal, and accident condition loads. The overpack is a large structure subject to a variety of complex loads and boundary conditions. The finite element model developed for this analysis allows efficient determination of the stresses in this complex structure.

The finite element model of the overpack is constructed using ANSYS [3.4.1]. This model is duplicated in the HI-STAR 100 SAR (10CFR71) submittal for transport.

For structural analysis purposes, the overpack is assumed to be symmetric about a diametral mid-plane. This assumption is reasonable because the purpose of the model is to investigate global stresses in the model. The model is not intended to resolve effects due to small penetrations that produce peak stresses (which are significant only in cyclic fatigue conditions).

Element plots of the model are shown in four figures (Figures 3.4.17 through 3.4.20). The basic building blocks of the finite element model are 20-node brick (SOLID95), 8-node brick (SOLID45), and 6-node tetrahedron elements (SOLID45). These are 3-D solid elements with 3 degrees of freedom at each node (three linear displacement degrees of freedom). Element densities are increased towards the top and bottom of the model in order to provide increased resolution of the stress fields in those regions.

The top flange/closure plate interface is modeled using linear spring elements (COMBIN14). The concentric seals are not modeled explicitly. The model is not intended to resolve the stress field around the grooves for the seals. The status of joint seal is ascertained by "compression springs" which simulate the O-ring gaskets. Contact between the overpack top flange and closure plate is verified by checking the status of these spring elements. If contact between the closure plate and top flange is maintained (indicated by a compressive load in the "compression springs"), then the integrity of the seal is determined to have been maintained.

The overpack closure bolts are modeled with beam elements (BEAM4). The top of the beam elements represents the bolt head and is connected to the overpack closure plate. The bottom of the elements represents the threaded region of the bolt and is connected to nodes of elements representing the top flange.

The inner shell of the overpack is modeled with two element layers through the thickness of the shell.

Each of the lifting trunnions is modeled as three rigid beam elements (BEAM4) connected to the top flange. The beams extend from the flange and meet at a single node location. Trunnion stress analysis is carried out in Supplement 20 of [3.4.13]; the inclusion of the trunnion herein is solely to provide the appropriate offset for handling loads.

The neutron shield material is not a load bearing or supporting component in the finite element model. However, the weight of the neutron shield material must be included in the model in order to obtain the proper inertia loads. The neutron shield material is modeled with SOLID45 elements having a weight density that is specified in Subsection 3.3.2.1. In the model herein, we include the neutron shield material as an element set to ensure that proper accounting of total weight (and accompanying deceleration loads) occurs. Therefore, the neutron shield material must be assigned a Young's Modulus in the model. A value approximately equal to 1% of the Modulus of the steel load carrying components is assigned to the neutron shield material to insure that the neutron shield material serves as a load rather than a structural member in the model.

It is recognized that the layered shells of the overpack are connected to each other and to innermost shell only at their top and bottom extremities. The finite element model must incorporate the potential for separation between the intermediate shells in certain regions under certain loading. Likewise, the intermediate shells cannot interpenetrate each other or the inner shell structure. This is accomplished by radially coupling adjacent intermediate shell nodes over two 60-degree spans. Figure 3.4.33 illustrates the nodal coupling pattern. The intermediate shell nodes that lie in the 60-degree sector between the top and bottom portions of the model remain uncoupled. The intermediate shells, in the uncoupled region, are free to separate from one another as the overpack cross section ovalizes during side impact. This modeling approach ensures that load transfer in a side drop is modeled correctly. With respect to the overpack model, "bottom portion" refers to the 60-degree segment of the model closest to the point of impact. Conversely, "top portion" refers to the 60-degree sector farthest from the point of impact. This nodal coupling arrangement conservatively represents the structural contribution of the intermediate shells. In addition, no axial or circumferential nodal coupling has been used between adjacent intermediate shells. Thus, axial bending stiffness of the composite shell structure is conservatively underestimated.

The two pocket trunnions (optional) at the base of the HI-STAR 100 overpack may be used for rotating the overpack from horizontal to vertical orientation (upending) or downending. In its role as a rotation pivot, the pocket trunnion is subject to minor stress levels as described in Subsection 3.4.3.5. Because the pocket trunnions are essentially local inserts in the form of a machined block in the layered shell structure, they do not provide a primary load transmission path within the HI-STAR 100 overpack. Therefore, the pocket trunnion is categorized as a non-Code part and in the finite element analysis of the overpack for storage; the pocket trunnions are not included in the structural representation of the overpack.

A one-dimensional force equilibrium evaluation has been performed to demonstrate that, even under the limiting accident condition of vertical drop leading to an inertial design basis deceleration of 60 g's, the structural joints between the outer layers of the overpack multi-shell structure and the cask body will maintain their structural integrity.

The overpack outer (5th layer) intermediate shell, radial channels, enclosure panels, and the Holtite-A neutron shielding material (total weight approximately 38,000 lbs. (169 kN)) are joined by a full-penetration circumferential weld to the top flange and a circumferential partial penetration groove weld to the fourth intermediate shell just above the bottom forging. The total axial load capacity of this weld configuration is greater than 22,000,000 lbs (98,000 kN). In the most limiting design basis scenario for storage (an end-drop handling accident), the amplified load on these welds is 2.28 million pounds (10,140 kN) (38,000 lb (169 kN) weight amplified by 60 g's). Sixty g's is the design basis deceleration limit for the HI-STAR 100 storage system. A comparison of weld load capacity against the amplified load indicates a safety factor greater than

9.9 for the weld. This provides assurance that the outer intermediate shell layer, the radial channels, and the neutron shield material will remain in place and perform their design function under all conditions of storage.

Elements at locations of welds in the modeled components are assumed to have complete connectivity in all directions. Material in the model located at positions where welds exist is assumed to have material properties identical to the base material.

To summarize, the total number of nodes and elements in the overpack model are 11265 and 8642, respectively. The elements used are SOLID45, SOLID95, BEAM4, SHELL63, and COMBIN14.

For all structural analyses, material properties are obtained from the appropriate tables in Section 3.3. Property data for temperatures that are not listed in the material property tables are obtained by linear interpolation. Property values are not extrapolated beyond the limits of the code for any structural analysis.

- Description of Individual Loads and Boundary Conditions

The method of applying each individual load to the overpack model is described in this subsection. The individual loads are listed in Table 2.2.14. A free-body diagram of the overpack corresponding to each individual load is given in Figures 3.4.21 through 3.4.26. In the following discussion, references to vertical and horizontal orientations are made. Vertical refers to the direction along the cask axis, and horizontal refers to a radial direction.

Quasi-static methods of structural analysis are used. The effects of any dynamic load factors (DLF) are discussed in the final evaluation of safety factors.

(a) Accelerations (Used to Form Load Cases 04.a and 04.b in Table 3.1.5)

Table 3.1.2 provides the bounding values of the accelerations used for design basis structural evaluation. The loading is imposed by amplifying the gravity vector by the design basis deceleration.

Boundary conditions for the model are as follows:

i. End drop - In an end drop, displacement fixities are applied to the model on a cross-section through the top flange that is normal to the drop direction. Figure 3.4.21 shows the free-body diagram for this load event. No reactions or internal body forces are shown.

ii. Side drop - In a side drop, the impacted region of the enclosure shell, radial channels, enclosure panels, and neutron shield located between the overpack and the impacting surface may sustain plastic deformation. Using a linear elastic overpack model, we cannot account for this behavior. For conservatism, the displacement constraints are placed directly at the outermost intermediate shell. That is, it is assumed that the outer radial plates and the outer enclosure have been rendered ineffective. The constraints are applied over an arc of 9 degrees. Figure 3.4.22 shows the free-body diagram. No reaction forces or internal body forces are shown.

(b) Loads on the Overpack from the MPC

Pressures are applied on the inner surfaces of the overpack model to represent loads from the MPC for the drop loads.

i. End drop - For a bottom end drop (Load Case 04.a in Table 3.1.5), the pressure load on the inside surface of the overpack bottom plate is assumed to be uniform and represents the load from the heaviest MPC (Figure 3.4.21). Note that this analysis conservatively assumes that the drop angle is not exactly 90º from the horizontal; attention is focussed on the overpack baseplate subject to the deceleration load from the heaviest MPC (applied as a uniform pressure) without the ameliorating effect of opposing distributed reaction from the impacted surface.

The magnitude of the pressure is the weight of the heaviest fully loaded MPC divided by the area of the faces of the elements over which the pressure is applied. The weight of the heaviest fully loaded MPC is taken from the tables in Section 3.2, and is amplified by the design basis deceleration.

ii. Side drop - The shape and extent of the pressure distribution is determined from the results of the structural analysis of the MPC presented in Subsection 3.4.4.3.1. In the MPC structural analysis, the extent of the support conditions of the MPC shell is determined with contact elements. The overpack is assumed to be a rigid circular surface. Based on the results of the MPC evaluations, the loaded region is taken as 81 degrees (measured from the vertical) and is applied as a sinusoid with maximum value at the line of symmetry.

The MPC load on the overpack model is applied uniformly along the axial length of the inner surface of the model. Figure 3.4.22 shows the overpack loading for the side drop event.

(c) Temperature (Used to Form Load Case 05 in Table 3.1.5)

Based on the results of the thermal evaluation for normal hot environment presented in Chapter 4, a temperature distribution with a bounding gradient is applied to the overpack model. The purpose is to determine the stress intensities that develop in the overpack under the applied thermal load. A plot of the applied temperature distribution as a function of radius is shown in Figure 3.4.2.

The temperature distribution is applied to the ANSYS finite element model at discrete nodes using a parabolic curve fit of the computed distribution. Figure 3.4.23 shows the displacement constraints for the thermal load case.

(d) Internal Pressure (Used to Form Load Cases 01 to 04 in Table 3.1.5)

Design internal pressure is applied to the overpack model. All interior overpack surfaces, including the inner shell, the bottom of the closure plate, and the top of the bottom plate are loaded with pressure. The magnitude of the internal pressure applied to the model is taken from Table 2.2.1. Figure 3.4.24 shows the displacement constraints for this load case.

(e) External Pressure (Used to Form Load Cases 01 to 04 in Table 3.1.5)

Design external pressure is applied to the overpack model. External pressure is applied to the model as a uniform pressure on the outer surface of the model. The magnitude of the external pressure applied to the model is taken from Table 2.2.1. Figure 3.4.25 shows the displacement constraints for this load case.

(f) Dead Weight in Vertical Orientation (Used in Load Cases 02 to 04 in Table 3.1.5)

A pressure load is applied on the top surface of the overpack bottom plate to represent the weight of the MPC. The magnitude of the pressure is the weight of the heaviest fully loaded MPC divided by the area of the faces of the elements over which the pressure is applied. The weight of the heaviest fully loaded MPC is taken from the tables in Section 3.2. The dead weight of the overpack itself is simulated by a 1g acceleration load in the appropriate direction. Figure 3.4.26 shows the displacement constraints for this load case.

(g) Handling Load (Used in Load Case 03 in Table 3.1.5)

As discussed in Section 3.1.2.1.2, a fully loaded HI-STAR 100 System using the lifting trunnions is a governing normal handling event. This load case is performed to determine the effects of normal condition handling loads on the overall overpack structure. This load case is intended to resolve the detailed stress distribution in the top flange in the region of the trunnion, to demonstrate that the ASME Code requirements (Level A Condition limits) are met.

Nodes in the region of the trunnions are fixed in all translational degrees-of- freedom. For additional conservatism, a vertical load amplifier of 2.0 is applied in this case (even though, as discussed in Subsection 3.4.3, the appropriate multiplier for a heavy load lift condition is 1.15).

A pressure load, applied on the top surface of the overpack bottom plate, represents the weight of the MPC. The magnitude of the pressure is the weight of the heaviest fully loaded MPC, amplified by 2.0, divided by the area of the faces of the elements over which the pressure is applied. Figure 3.4.26 shows the displacement constraints for this load case.

(h) Bolt Pre-load (Used in Load Cases 01-05 in Table 3.1.5)

The overpack closure bolts are torqued to values given in Chapter 8. This torque generates a pre-load in the bolts and stresses in the closure plate and top flange in the region adjacent to the bolts. This load is applied to the overpack model by applying an initial strain to the beam elements representing the bolts.

- Finite Element Analysis Solution Procedure

The analysis procedure is as follows:

1. The stress and deformation field due to each individual load is determined.

2. The results for each individual load are combined in a postprocessor to create each load case. The load cases analyzed are listed in Table 3.1.5.

3. The results for each load case are compared to allowables. The calculated values are compared with allowable values in Subsection 3.4.4.4.

3.4.4.3.2.2 Fabrication Stress

The fabrication stresses originate from welding operations to affix the intermediate shells in position. As the molten weld metal solidifies, it shrinks pulling the two parts of the shells together. Adjacent points at the weld location will close together after welding by an amount δ which is a complex function of the root opening, shape of the bevel, type of weld process, etc. The residual stresses generated by the welding process are largely confined to the weld metal and the "heat affected zone". The ASME Code recognizes the presence of residual stresses in the welds, but does not require their calculation. The Code also seeks to minimize fabrication stresses in the welds through controlled weld procedures. Nevertheless, fabrication stresses cannot be eliminated completely. Similarly, Regulatory Guide 3.61 does not require computation of stresses arising from the manufacturing operations.

The computation of fabrication stresses, however, is carried out to comply with the provisions of Regulatory Guide 7.8, Article C-15 when the HI-STAR 100 is functioning as a transport cask. The Regulatory Guide requires that "Fabrication and installation stresses in evaluating transportation loadings should be consistent with the joining, forming, fitting, and aligning processes employed during the construction of casks...the phrase fabrication stresses includes the stresses caused by interference fits and the shrinkage of bonded lead shielding during solidification but does not include the residual stresses due to plate formation, welding, etc.".

A literal interpretation of the above-cited Regulatory Guide text exempts the HI-STAR 100 designer from computing the stresses in the containment shell due to welding. However, in the interest of conservatism, it was decided to compute and establish an upper bound on the stresses induced in the containment shell ("helium retention boundary" in the storage mode) and in the intermediate shells due to welding of the intermediate shell layers. Detailed calculations are presented in Supplement 24 of [3.4.13].

To calculate the so-called fabrication stresses, we recall that in affixing the intermediate shells to the cask body, the design objective does not call for a definite radial surface pressure between the layers. Rather, the objective is to ensure that the shells are not loosely installed. Fortunately, extensive experience in fabricating multi-layer shells has been acquired by the industry over the past half-century. The technology that was developed and has matured for fabrication in older industries (such as oil and chemical) will be used in HI-STAR 100 fabrication of the multi-layered shells. Mock-up tests on carbon steel coupons indicate that the total shrinkage after welding can range from 0.010" (0.254 mm) to 0.0625" (1.59 mm) for the bevel and fit-up geometry in the HI-STAR 100 design drawings. Therefore, the calculations in Supplement 24 of [3.4.13] are carried out using the upper bound gap of 0.0625" (1.59 mm). To bound the computed stresses even further, the inter-layer friction coefficient is set equal to zero. It is intuitively apparent that increasing the friction increases the localized stresses near the "point of pull" (i.e., the weld) while mitigating the stresses elsewhere. Since our object is to maximize the distributed (membrane) stress, the friction coefficient is set equal to zero in the analysis of Supplement 24 of [3.4.13].

The results from the analyses in Supplement 24 of [3.4.13] are summarized in Table 3.4.16.

The results shown in the table leads to the conclusion that the maximum possible values for stresses resulting from the HI-STAR 100 fabrication process are only a fraction of the relevant ASME Code limit.

3.4.4.3.2.3 Structural Analysis of Overpack Closure Bolting (Load Cases 01, 06 and 04.a - Table 3.1.5)

Stresses are developed in the closure bolts due to pre-load, pressure loads, temperature loads, and accident loads. Closure bolts are explored in detail in Reference [3.4.5] that was prepared for analysis of shipping casks. The method presented is equally valid for storage casks and is considered as an acceptable analysis method by NUREG-1536. The analysis of the overpack closure bolts under normal and accident conditions appropriate for storage is carried out in Supplement 21 of [3.4.13] and follows the procedures defined in Reference [3.4.5]. The allowable stresses used for the closure bolts follow that reference.

The following combined load cases are analyzed in Supplement 21 of [3.4.13].

Normal: pressure, temperature, and pre-load loads are included. (Load Case 01 in Table 3.1.5).

Top Closure Puncture: pressure, temperature, pre-load, and 8-in (203 mm) diameter missile loads are included. (Load Case 06 in Table 3.1.5)

Drop: pressure, temperature, pre-load, and impact loads from a top end drop are included. We note that reference [3.4.5] is for shipping casks and therefore allows for a top end drop. There is no such credible event defined for a storage cask but it provides a bounding case for the HI-STAR 100 closure bolts (Load Case 04.a in Table 3.1.5).

Reference [3.4.5] reports safety factors defined as the calculated stress combination divided by the allowable stress for the load combination. This definition of safety factor is the inverse of the definition consistently used in this FSAR. In summarizing the closure bolt analyses performed in Supplement 21 of [3.4.13], we report results using the safety factor definition of allowable stress divided by calculated stress. The results for closure lid bolting are obtained from Supplement 21 of [3.4.13] and are summarized in Table 3.4.17.

It is seen from the table that all safety factors are greater than 1.0.

3.4.4.3.2.4 Structural Analysis of the Overpack Closure Plate (Load Case 04.a in Table 3.5.1)

The simplified analysis given here complements the result from the finite element analysis of the overpack for calculation of stresses in the overpack closure plate.

The loading condition considered here is a bottom end drop where the overpack bottom plate impacts the supporting surface (Load Case 04.a in Table 3.1.5).

The following assumptions apply:

1. Stresses in the closure plate due to bolt pre-load will counteract the downward inertia load; the pre-load is conservatively ignored.
2. The closure plate is assumed to be a simply supported plate, i.e., the rotational fixity projected by the flanged joint is conservatively ignored.
3. The plate is assumed to be loaded with its own weight multiplied by the design basis deceleration from Table 3.1.2.
4. The pressure within the overpack counteracts the amplified self-weight load in a bottom end drop. Internal overpack pressure is conservatively neglected.

The geometry of the model is the same as shown in Figure 3.4.16 for the MPC lid except for the dimension change appropriate to the overpack closure plate. Using Table 24, Case 10 of reference [3.G.1], page 429 (reference is listed in Supplement 22 of [3.4.13]), the maximum radial bending stress (σ) in the closure plate due to bending is

$$\sigma = \frac{3qa^2(3+\nu)}{8t^2}$$

where

q = load per unit area = $\frac{a_v W}{\pi a^2}$

W = weight of the plate = 8,000 lb (36 kN).

a_v = design basis deceleration in an end drop = 60g's

a = radius of the simply supported plate = 38.6875 in (982.7 mm)

υ = Poisson's ratio

t = thickness of the closure plate = 6 in (152.4 mm)

Therefore,

$$\sigma = \frac{3(3+\nu)W\, a_v}{8\,\pi\, t^2}$$

The result is summarized in Table 3.4.18 and the safety factor is much greater than 1.0.

3.4.4.3.2.5 Elastic/Plastic Stability Considerations for the Overpack Inner Shell (Load Cases 02, 03 and 04.a in Table 3.1.5)

Supplement 23 of [3.4.13] contains a complete stability analysis of the HI-STAR 100 System. The case of normal handling (Load Case 03 in Table 3.1.5), the accident end drop (Load Case 04.a in Table 3.1.5), and the accident external pressure plus dead load case (Load Case 02 in Table 3.1.5) are evaluated for elastic and plastic stability in accordance with the ASME Code Case N-284 [3.4.6]. All required interaction equation requirements set by [3.4.6] are met. It is shown in Supplement 23 of [3.4.13] that yield strength limits rather than instability limits governs the minimum safety factor. Minimum safety factors from Supplement 23 of [3.4.13] are summarized in Table 3.4.19.

3.4.4.3.2.6 Stress Analysis of Enclosure Shell

The overpack enclosure shell and the overpack enclosure return are examined for structural integrity under a bounding internal pressure in Supplement 30 of [3.4.13]. It is shown there that large safety factors exist against overstress due to an internal pressure developing from off-gassing of the neutron absorber material.

In HI-STAR 100 serial numbers 1 through 4, the enclosure shell is lined with a thin layer of thermal expansion foam. The purpose of the foam is to mitigate the pressure on the enclosure shell caused by radial expansion of the neutron absorber material at high temperatures.

As of July 2000, the radially oriented foam was removed from the HI-STAR 100 design. A finite element analysis has been performed to demonstrate that the removal of this foam material does not lead to excessive stress levels in the enclosure shell. Figure 3.4.46 shows a picture of the 2-D finite element model, which was created using ANSYS Version 5.4. Two-dimensional plate elements are used to model the steel channels and the neutron absorber material. The interface between the two materials is simulated by contact elements. The maximum stress in the radial channel under the worst-case thermal loading is 46,251 psi (319 MPa), which is less than the ultimate stress of the material at 300°F (70,000 psi) (149°C (483 MPa)). Therefore, the removal of the radially directed expansion foam is justified.

3.4.4.4 Comparison with Allowable Stresses

Consistent with the formatting guidelines of Reg. Guide 3.61, calculated stresses and stress intensities from the finite element and classical elasticity evaluations are compared with the allowable stresses and stress intensities defined in Subsection 3.1.2.2 per the applicable service conditions and the ASME Code relevant for the component. Safety factors for those components that are identified as lifting devices have been reported in Subsection 3.4.3.

3.4.4.4.1 MPC Fuel Basket and Enclosure Vessel

It is recalled that the stress analyses for the load cases applicable to the fuel basket and the enclosure vessel (EV) (that together constitute the Multi-Purpose Canister) are stated in Tables 3.1.3 and 3.1.4, respectively. All detailed analyses, including finite element model details and the necessary explanations to collate and interpret the voluminous numerical results are contained in [3.4.13] and [3.4.14]. These analyses are identified in Tables 3.1.3 and 3.1.4 for ease of reference. Summaries of results for the load cases pertinent to the fuel basket and the enclosure vessel (EV) are provided in Tables 3.4.7-3.4.9 wherein the source analyses containing detailed results are also identified. To further facilitate perusal of results, another level of summarization is performed in Tables 3.4.3-3.4.4 where the global minima of safety factor for each load case are presented. The following elements of information are relevant in ascertaining the safety factors under the various load cases presented in the tables.

- In the interest of simplification of presentation and conservatism, the total stress intensities under mechanical loading are considered to be of the primary genre' even though, strictly speaking, a portion can be categorized as secondary (that have much higher stress limits).
- In load cases involving accident events, the deceleration loads also produce internal dynamic effects, qualified as dynamic load factors (DLF). These DLF's, which depend on the duration of impact and the fundamental frequency of the internal component (e.g. the fuel basket panel) are computed in Supplement 25 of [3.4.13] using input data from Supplement 18 of [3.4.13]. The factors of safety presented in the above mentioned summary tables do not include the DLF's. Therefore, the tabulated factors of safety should be compared against the applicable DLF to insure that a positive design margin exists for a given load case.

The MPC stress distributions that correspond to the worst case for each MPC are provided in Figures 3.4.34 and 3.4.36. The stresses appear as colored bars where the height is proportional to the magnitude of the stress and the width equals the element length. The figures also include the design temperature of the component and the allowable stress intensity taken from Table 3.1.17 for the Level D condition of primary membrane plus bending stress. We note that for the MPC-68, the worst case is in the basket support structure.

A perusal of the results for Tables 3.4.3 and 3.4.4 under different load combinations for the fuel basket and the enclosure vessel reveals that all factors of safety are above 1.0 even if we use the most conservative value for dynamic amplification factor. The relatively modest factor of safety in the fuel basket under side drop events (Load Case F3.b and F3.c) in Table 3.4.3 warrants further explanation.

The wall thickness of the storage cells, which is by far the most significant variable in the fuel basket's structural strength, is significantly greater in the HI-STAR 100 MPCs than in comparable fuel baskets licensed in the past. For example, the cell wall thickness in the TN-32 basket (Docket No. 72-1021, M-56) is 0.1 inch (2.54 mm) and that in the NAC-STC basket (Docket No. 71-7235) is 0.048 inch (1.22 mm). In contrast, the cell wall thickness in the MPC-24 is 0.3125 inch (7.94 mm). In spite of their relatively high flexural rigidities, computed margins in the HI-STAR 100 fuel baskets are rather modest. This is because of some assumptions in the analysis which lead to an overstatement of the state of stress in the fuel basket. For example:

i. The section properties of longitudinal fillet welds that attach contiguous cell walls to each other are completely neglected in the finite element model (Figure 3.4.7). The fillet welds strengthen the cell wall section modulus at the very locations where maximum stresses develop.

ii. The radial gaps at the fuel basket-MPC shell and at the MPC shell-overpack interface are explicitly modeled. As the applied loading is incrementally increased, the MPC shell and fuel basket deform until a "rigid" backing surface of the overpack is contacted, making further unlimited deformation under lateral loading impossible. Therefore, some portion of the fuel basket and enclosure vessel (EV) stress has the characteristics of secondary stresses (which by definition, are self-limited by deformation in the structure to achieve compatibility). For conservativeness in the incremental analysis, we make no distinction between deformation controlled (secondary) stress and load controlled (primary) stress in the stress categorization of the MPC-32 and 68 fuel baskets. We treat all stresses, regardless of their origin, as primary stresses. Such a conservative interpretation of the Code has a direct (adverse) effect on the computed safety factors. As noted earlier, the results for the MPC-24 are properly based on primary stresses to illustrate the conservatism in the reporting of results for the MPC-32, 68 baskets.

iii. The SNF inertia loading on the cell panels is simulated by a uniform pressure, which is a most conservative approach for incorporating the SNF/cell wall structure interaction.

The above assumptions act to depress the computed values of factors of safety in the fuel basket finite element analysis and render conservative results.

As stated earlier, the reported values do not include the effect of dynamic load amplifiers. As noted in Appendix 3.A and Supplement 25 of [3.4.13], the duration of impact and the predominant natural frequency of the basket panels under lateral drop events, result in dynamic load factors (DLF) below 1.1. Therefore, since all reported factors of safety for the fuel basket panels (based on stress analysis) are greater than the DLF, the MPC fuel basket is structurally adequate for its intended functions during and after a lateral drop event.

Tables 3.4.7 and 3.4.8 report stress intensities and safety factors for the confinement boundary subject to internal pressure alone and internal pressure plus the normal operating condition temperature with the most severe thermal gradient. The final values for safety factors in the various locations of the confinement boundary provide assurance that the MPC enclosure vessel is a robust pressure vessel.

3.4.4.4.2 Overpack

3.4.4.4.2.1 Discussion

The overpack is subject to the load cases listed in Table 3.1.5. Results from the series of finite element analyses are summarized in Table 3.4.10. In order to identify and to locate appropriate regions with limiting safety factors, we note that safety factors are calculated for each load case at a select set of nodes identified as "stress report locations".

Figures 3.4.37 through 3.4.42 identify the locations of minimum safety margin for all overpack load combinations.

For any load case associated with a Level D Service Condition (accident event), the results are provided only at locations where primary stresses predominate. However, in the interest of conservatism, the stress intensity reported at these locations does not separate out any secondary components. Every value is placed in the primary membrane or primary bending category, even though many locations clearly involve non-primary bending components. This simplification in stress intensity tabulation imputes considerable additional safety margin to the processed results, which is not explicitly recognized in the results presented herein.

The following text is a brief description of how the results are presented for evaluation:

- The extensive body of results is summarized in Table 3.10 wherein the minimum safety factor for different components of the overpack for each of the load cases is presented.

- Table 3.4.6 presents results of calculation of the safety factors to include “fabrication stresses” where appropriate. Table 3.4.6 summarizes safety factors, based on limits for primary plus secondary stresses, and reports the limiting safety factors for the overpack shells for events subject to Level A Service Conditions. Fabrication stresses are not included for any load case involving accident conditions since secondary stresses need not be considered for Level D Service Conditions. Fabrication stress, reported in Subsection 3.4.4.3.2.2, is “added” in absolute value to finite element stress intensity or stress results. This conservatively produces modified stress intensity or stress that is used to compute modified safety factors.

- Finally, Table 3.4.5 summarizes the minimum values of safety factors (global minima) for the overpack components.

The modifications summarized in Table 3.4.6 are briefly discussed below:

Case 01 (Pressure) –Safety factors are summarized in Table.3.4.10 prior to inclusion of fabrication stress. Table 3.4.6 shows modified safety factors that include fabrication stress. The pressure stresses result in tensile longitudinal and circumferential stresses in the inner shell and in the intermediate shells. The fabrication stress dominates the stress state in the inner and intermediate shells but comparison with the allowable values is considers a primary plus secondary stress state.

Case 03 (Normal Handling): Safety factors are summarized in Table.3.4.10 prior to inclusion of fabrication stress. Table 3.4.6 shows modified safety factors that conservatively include fabrication stress but compute safety factors considering primary plus secondary stress allowables.

Case 05 (Thermal Load) - Safety factors are summarized in Table.3.4.10 prior to inclusion of fabrication stress. Table 3.4.6 shows modified safety factors that include the effect of fabrication stress in a conservative manner. Safety factors are based on allowable strengths for primary plus secondary stresses since thermal stress is a secondary stress.

3.4.4.4.3 Result Summary for the Heat Condition

- Stress Results from Overall Finite Element Models of the MPC and Overpack

Tables 3.4.7 to 3.4.10 summarize minimum safety factors from load cases analyzed using the finite element models of the MPC fuel basket plus canister and the overpack described in Subsections 3.4.4.3.1.1 and 3.4.4.3.2. All safety factors are greater than 1.0 and are greater than any credible dynamic amplifier for the location. Table 3.4.6 provides a summary table that includes the effect of fabrication stress on safety factors for the intermediate and inner shells of the overpack. Table 3.4.6 reports safety factors based on primary plus secondary allowable strengths.

- Status of Lid Bolts and Seals on the Overpack

The finite element analysis for the overpack provides results at the lid-to top flange interface. The output results for each load combination indicate that all seal springs remain closed under load indicating that the sealworthiness of the bolted joint will not be breached.

For each load combination, the total compressive force on the closure plate-overpack interface as well as the total tangential force (labeled as "friction force" in the tables) is computed. If the ratio "total friction force/total compressive force" is formed for each set of results, the maximum value of the ratio is 0.268. There will be no slip of the closure plate relative to the overpack if the interface coefficient of friction is greater than the value given above. Note that Mark's Handbook for Mechanical Engineers [3.4.9] in Table 3.2.1 shows $\mu = 0.74$-0.79 for clean and dry steel on steel surfaces. It is concluded that there is no propensity for relative movement.

Based on the results of the finite element analysis, the following conclusions are made.

No bolt overstress is indicated under any loading event. Note that this confirms the results of closure bolt analyses performed in Supplement 21 of [3.4.13].

The closure plate seals do not unload under any load combination; therefore, the seals continue to perform their function.

- Stress and Stability Results from Miscellaneous Component Analyses in Subsection 3.4.4.3

Tables 3.4.11 to 3.4.19 provide summary results from additional analyses described and reported on in Subsection 3.4.4.3 for components of the MPC and the overpack. The tables have been referred within the text of Subsection 3.4.4.3. The tables report comparisons of calculated values with allowable values for both stress and stability and represent a compilation of analyses detailed in [3.4.13] and [3.4.14].

- Summary of Minimum Safety Factors

Tables 3.4.3 through 3.4.5 present a concise summary of safety factors for the fuel basket, the enclosure vessel, and the overpack, respectively. Locations in this FSAR where detailed information on each summary value exists is also identified in these tables.

Based on the results of all analyses, with results presented or summarized in the text, in tabular form, and in [3.4.13] and [3.4.14], we close by concluding that:

i. All safety factors reported in the text, summary tables, and in appendices are greater than 1.0.

ii. There is no restraint of free thermal expansion between component parts of the HI-STAR 100 System.

3.4.5 Cold

A discussion of the resistance to failure due to brittle fracture is provided in Subsection 3.1.2.3.

The value of the ambient temperature has two principal effects on the HI-STAR 100 storage system, namely:

i. The steady-state temperature of all material points in the cask system will go up or down by the amount of change in the ambient temperature.

ii. As the ambient temperature drops, the absolute temperature of the contained helium will drop accordingly, producing a proportional reduction in the internal pressure in accordance with the Ideal Gas Law.

In other words, the temperature gradients in the cask system under steady-state conditions will remain the same regardless of the value of the ambient temperature. The internal pressure, on the other hand, will decline with the lowering of the ambient temperature. Since the stresses under normal storage condition arise principally from pressure and thermal gradients, it follows that the stress field in the MPC under -40ºF (-40°C) ambient would be smaller than the "heat" condition of storage, treated in the preceding subsection. Therefore, the stress margins computed in Section 3.4.4 can be conservatively assumed to apply to the "cold" condition as well. Supplement 28 of [3.4.13] demonstrates that the overpack closure bolts will retain the helium seal under the cold ambient conditions.

Under the 80°F (27°C) ambient temperature and the maximum fuel decay heat load (normal heat condition of storage), the thermal analysis in Chapter 4 reports the resultant component temperatures. These temperatures were than used in calculations to demonstrate that there was no restraint of free thermal expansion for the MPC-24, 32, and 68 in the HI-STAR overpack. The results from these calculations have been presented in Subsection 3.4.4.2.1. Under the postulated cold ambient temperature of -40°F (-40°C) the component temperatures will decrease by 80°F (27°C) minus -40°F (-40°C) or 120°F (49°C). Thermal expansion is calculated from the product of the coefficient of thermal expansion, α, and the change in temperature, ΔT. Since the changes in temperature in each component would decrease by 120°F (49°C), the resultant thermal expansion would also decrease. This is coupled with the fact that the coefficient of thermal expansion for carbon steel and stainless steel decreases as the temperatures are decreased. Therefore, if the analysis performed under hot conditions demonstrate that there is no restraint of thermal expansion, analysis performed at component temperatures 120°F (49°C) less (to account for the cold ambient temperature, -40°F (-40°C) would also show that there is no constraint of thermal expansion. That is, the operational clearances predicted in Subsection 3.4.4.2.1 are a conservative lower bound on the clearances with the ambient temperature corresponding to extreme cold conditions.

Finally, the HI-STAR 100 System is engineered to withstand “cold” temperatures (-40ºF) (-40°C) without impairment of its storage function.

The structural material used in the MPC (Alloy X) and the Helium Retention Boundary is recognized to be completely immune from brittle fracture in the ASME Codes. As no liquids are included in the HI-STAR 100 overpack design, loads due to expansion of freezing liquids are not considered.

3.4.6 HI-STAR 100 Kinematic Stability under Flood Condition (Load Case A in Table 3.1.1)

The flood condition subjects the HI-STAR 100 System to external pressure, together with a horizontal load due to water velocity.

The design external pressure bounds any credible pressure due to complete submergence during flooding (see Subsection 3.1.2.1.1.3 and Supplement 23 of [3.4.13]). The use of such a large external design pressure is mandated by 10CFR71 considerations.

Horizontal loads on the HI-STAR 100 System may, however, cause sliding (translation), or rotation (tipping); this is addressed below, where it is shown that the maximum permitted flood water velocity is limited by sliding of the cask.

Rotation of the HI-STAR 100 System due to motion of the floodwater is analyzed by assuming that the overpack is pinned at the outer edge of the baseplate opposite the water flow. The pinned edge does not permit sliding.

The water velocity associated with flood produces a horizontal drag force, which may act to cause sliding or tip-over. In accordance with the provisions of ANSI/ANS 57.9, the acceptable upper bound flood velocity, V, must provide a minimum factor of safety of 1.1 against overturning and sliding. For HI-STAR 100, the design basis flood velocity is 13 feet/sec.

The overturning horizontal force, F, due to hydraulic drag, is given by the classical formula:

$$F = Cd\ A\ V^*$$

where:

V^* is the velocity head = $\frac{\rho V^2}{2g}$; (ρ is water weight density, g is acceleration due to gravity, and V is the crossflow water velocity).

A: projected area of the HI-STAR 100 cylinder perpendicular to the fluid velocity vector.

Cd: drag coefficient

The value of Cd for flow past a cylinder at Reynolds number above 5E+05 is given as 0.5 in the literature (viz. Hoerner, Fluid Dynamics, 1965).

The drag force tending to cause HI-STAR 100's sliding is opposed by the friction force, which is given by

$$F_f = \mu\ K\ W$$

where:

μ = limiting value of the friction coefficient at the HI-STAR 100/ISFSI pad interface (conservatively taken as 0.25, although literature citations give somewhat higher values).

K = buoyancy coefficient

W= Minimum weight of HI-STAR 100 with an empty MPC

Sliding Factor of Safety

The factor of safety against sliding, SF_1, is given by

$$SF_1 = \frac{F_f}{F} = \frac{\mu KW}{Cd\,A\,V^*}$$

It is apparent from the above equation that SF_1 will be minimized if the lower bound weight of HI-STAR 100 is used in the above equation.

As stated previously, $\mu = 0.25$, $Cd = 0.5$.

V^* corresponding to 13 ft/sec (3.96 m/s) water velocity (see Chapter 2 Design Criterion) is 163.75 lb per sq. ft (7,840 Pa).

A = length x diameter of HI-STAR 100 = 96" x 203"/144 sq. in/sq. ft = 135.33 sq. ft (12.57 sq. m)

W = 189,000 lbs. (841 kN) (Table 3.2.1)

K = buoyancy factor = (1-weight of water displaced by HI-STAR 100/W) = 0.719

Therefore, the drag force is

$F_f = \mu\,K\,W$ = 33,973 lb. (151 kN)

Substituting in the above formula for SF_1, we have

$SF_1 = 3.07 > 1.1$ (required)

Overturning Factor of Safety

For determining the margin of safety against overturning SF_2, the cask is assumed to pivot about a fixed point located at the outer edge of the contact circle at the interface between HI-STAR 100 and the ISFSI. The overturning moment due to a force F_T applied at height H^* is balanced by a restoring moment from the reaction to the cask buoyant force KW acting at radius D/2.

$$F_T\,H^* = KW\frac{D}{2}$$

or

$$F_T = \frac{K\,W\,D}{2\,H^*}$$

W is the minimum weight of the storage overpack with an empty MPC.

We have,

W = 189,000 lb (841 kN) (Table 3.2.1)

H^* = 103.2" (2,621 mm) (maximum height of mass center per Table 3.2.2)

D = 83.25" (2,115 mm) (Holtec Drawing 3913, Sheet 2)

K = 0.719 (calculated)

so that

F_T = 54,811 lb (244 kN)

F_T is the horizontal drag force at incipient tip-over.

F = Cd A V^* = 11,080 lbs (49.3 kN) (drag force at 13 feet/sec (3.96 m/s))

The safety factor against overturning, SF_2, is given by

$$SF_2 = \frac{F_T}{F} = 4.95 > 1.1 \text{ (required)}$$

The sliding and overturning evaluations performed above for the vertically oriented cask are also bounding for a loaded HI-STAR overpack stored on an ISFSI pad in the horizontal orientation for the following reasons.

The sliding evaluation depends on the mass and the projected (or impingement) area of the cask. These two quantities are quite similar whether the cask is stored horizontally or vertically. In the horizontal orientation, the cask rests on a low profile supporting structure, which adds slightly to the total mass and the impingement area, but these increases are offsetting with respect to the net sliding force. Therefore, the sliding evaluation presented above is also bounding for the horizontal orientation.

With regard to the overturning evaluation, the cask is inherently more stable when it is stored horizontally because of its lower overall height above the ISFSI pad[1] and the added mass of the low profile supporting structure, both of which reduce the net overturning moment on the cask due the moving floodwater.

3.4.7 Seismic Event on HI-STAR 100 (Load Case C in Table 3.1.1)

3.4.7.1 Stability

3.4.7.1.1 Stability of Vertically Arrayed HI-STAR 100

The HI-STAR 100 System plus its contents are subject to the design basis seismic event consisting of three orthogonal statistically independent acceleration time-histories (orthogonal

[1] Center of gravity height (i.e., centerline) of the cask must not exceed 72” above the ISFSI pad surface.

components). The HI-STAR 100 System can be considered as a rigid body subject to a net horizontal inertia force and a vertical inertia force for the purpose of performing a conservative analysis to determine the maximum ZPA that will not cause incipient tipping. The vertical seismic loading is conservatively assumed to act in the most unfavorable direction (upwards) at the same instant. The vertical seismic load is assumed to be equal to or less than the net horizontal load with ε being the ratio of vertical component to one of the horizontal components. Define D as the contact patch diameter, and H_{CG} as the height of the centroid of an empty HI-STAR 100 System (no fuel).

D = 83.25" (2,115 mm) (Drawing 3913, Sheet 2)

Tables 3.2.1 and 3.2.2 give HI-STAR 100 weight data and center-of-gravity heights.

The weights and center-of-gravity heights are reproduced here for calculation of the composite center of gravity height of the overpack together with an empty MPC.

Weight (pounds) (kN)	H of C.G. Height (Inches) (mm)
Overpack - W_o = 153,710 (684)	99.7 (2,532)
MPC-24 - W_{24} = 40,868 (182)	109.0 + 6 = 115.0† (2,769+152 = 2,921)
MPC-32 - W_{32} = 34,507 (153)	113.2 + 6 = 119.2† (2,875+152 = 3,027)
MPC-68 - W_{68} = 37,591 (167)	111.5 + 6 = 117.5† (2,832+152 = 2,984)

† MPC centroids reported in Section 3.2 are measured from the base of the MPC.

The composite centroid, H_{CG}, is determined from the equation

$$H_{cg} = \frac{W_o \text{ x } 99.7 + W_{MPC} \text{ x } H}{W_o + W_{MPC}}$$

Performing the calculations for all of the MPC's gives the following results:

	H_{CG} (inches) (mm)
MPC-24 with overpack	102.9 (2,614)
MPC-32 with overpack	103.3 (2,624)
MPC-68 with overpack	103.2 (2,621)

A conservative stability limit is achieved by using the largest value of H_{CG} (call it H) from above or from Table 3.2.2. Because the HI-STAR 100 System is a radially symmetric structure, the two horizontal seismic accelerations can be combined vectorially and applied as an overturning force at the C.G. of the cask. The overturning static moment in each of the two horizontal directions is "WGH_{CG}" where W is the total system weight, G is the zero period acceleration seismic amplifier so that WG is the inertia load due to horizontal seismic loading. The overturning moment is balanced by a vertical reaction force, acting at the outermost contact patch radial location r = D/2. At many sites, the vertical seismic acceleration is specified as a fraction of the horizontal acceleration. Let us assume that the vertical acceleration is ε times G. The resistive moment is minimized when the vertical acceleration tends to reduce the apparent weight of the cask. At that instant, the moment that resists "incipient tipping" is:

$$W(1-\varepsilon G)\,r$$

where the vertical seismic amplifier is εG ($|\varepsilon| \leq 1$).

Equating the two moments to ensure equilibrium of moments yields

$$\sqrt{(WG)^2+(WG)^2}\,H = W\,(1-\varepsilon G)\,r$$

$$\sqrt{2}\,WGH = W\,(1-\varepsilon G)\,r$$

or, after canceling W and solving for G

$$G = \frac{1}{\sqrt{2}\,\frac{H}{r}+\varepsilon}$$

The values of r and H for the HI-STAR 100 are r = 41.625" (1,057 mm), H = 103.3" (2,624 mm), which yields the following results for different values of ε

Acceptable Horizontal g-Level in Each of Two Orthogonal Directions	Vertical Acceleration Multiplier (ε)	Vectorial Sum of Acceptable Horizontal Accelerations (g)
0.222	1.0	0.314
0.235	0.75	0.332
0.240	0.667	0.339
0.250	0.50	0.354

The tabular results above define the envelope g-levels from the resultant inertia load from two horizontal seismic events to ensure against incipient tipping. The acceptable g-level is increased as the ratio of vertical zero period accelerations to net horizontal g-level decreases. Additionally, in case of a 2-D earthquake plant, i.e., one horizontal and one vertical seismic acceleration, the acceptable g-level will correspond to the last two columns in the above table.

3.4.7.1.2 Stability of Horizontally Arrayed HI-STAR 100

The stability of the horizontally stored HI-STAR 100 System is determined following the same approach as vertically arrayed HI-STAR 100 System. During the horizontal storage, the HI-STAR 100 may be staged on cribbing or other supporting structures. Define H_{CGH} as the height from the centroid of an empty HI-STAR 100 System (no fuel) to the base of the supporting structure, and B as the width of the supporting structure. Because the dimension of HI-STAR 100 System is greater in height than diameter, the HI-STAR 100 System tends to overturn about the longitudinal axis. Therefore, the horizontal seismic accelerations do not need to be combined vectorially, and the maximum of two horizontal seismic accelerations is applied as an overturning force at the centroid height of the cask (H_{CGH}). The overturning moment is balanced by a vertical reaction force, acting at the outermost contact patch of the supporting structure location B. The overturning moment is:

$$WGH_{CGH}$$

The moment that resists "incipient tipping" is:

$$W\,(1-\varepsilon G)\,(B/2)$$

Equating the two moments to ensure equilibrium yields

$$WGH_{CGH} = W\,(1-\varepsilon G)\,(B/2)$$

and, after canceling W, the following relationship results

$$\frac{H_{CGH}}{B} \leq \frac{(1-\varepsilon\,\mathrm{G})}{2G}$$

where ε and G are defined in Subsection 3.4.7.1.1.

The HI-STAR 100 System is stable as long as the above inequality is satisfied. As an example, for H_{CGH} = 72" and B = 96", the following results are obtained for different values of ε.

Acceptable Horizontal g-Level in Each of Two Orthogonal Directions	Vertical Acceleration Multiplier (ε)	Vectorial Sum of Acceptable Horizontal Accelerations (g)
0.40	1.0	0.566
0.44	0.75	0.622
0.46	0.667	0.651
0.50	0.50	0.707

If the above inequality cannot be satisfied for a particular site, then a 3-D time history analysis may be performed to demonstrate the stability of the HI-STAR 100 overpack in its horizontal storage configuration.

Lastly, it is noted that the supporting structure that is used for horizontal storage of a loaded HI-STAR overpack is an ancillary device, which is classified as not important to safety (NITS). The justification for its NITS designation is because the HI-STAR 100 overpack has been designed and qualified to withstand a horizontal drop from a height of 72" above the ISFSI pad surface (see Appendix 3.A). Therefore, by requiring that the center of gravity height of the HI-STAR 100 overpack above the ISFSI pad be limited to 72" or less when stored horizontally, a gross failure of the supporting structure does not pose any safety risk to the HI-STAR 100 System.

3.4.7.2 Primary Stresses in the HI-STAR 100 Structure

A simplified calculation to assess the flexural bending stress in the HI-STAR 100 structure under the limiting seismic event (at which tipping is incipient) is presented in the following:

From the acceptable acceleration table presented above, the maximum horizontal acceleration is 0.354g. The corresponding lateral seismic load, F, is given by F = 0.354 W. This load will be maximized if the upper bound HI-STAR 100 weight (W = 245,000 lbs. (1,090 kN), from Table 3.2.4) is used. Accordingly, F = (0.354) (245,000) = 86,730 lbs (F = (0.354) (1,090) = 385.9 kN).

The moment, M, at the base of the HI-STAR 100 due to this lateral force is given by

$$M = \frac{F\,H}{2}$$

where H = height of HI-STAR 100 (taken conservatively as 204 inches (5.18 m))

The flexural stress, σ, is conservatively given by the ratio of the moment M to the section modulus of the inner steel shell structure, z, which is computed to be 9,644 in.^{3} (0.158 m^3).

Therefore,

$$\sigma = \frac{(86{,}730)\,(204)}{(9{,}644)\,(2)} = 917 \text{ psi} \qquad \left(\sigma = \frac{(385.9)\,(5.18)}{(0.158)\,(2)} = 6.32 MPa\right)$$

We note that the contribution from any of the intermediate shells has been neglected in the above calculation.

The maximum axial stress in the overpack shell will be reached in the "compressive" side where the flexural bending stress algebraically sums with the direct compression stress τ from vertical compression.

From the acceleration table the vertical seismic accelerations corresponding to the net 0.354g horizontal acceleration is 0.125g.

Therefore, using the maximum overpack weight

$$\tau = \frac{(245{,}000)\,(1.125)}{560} = 492 \text{ psi} \qquad \left(\tau = \frac{(1{,}090)\,(1.125)}{0.361} = 3.39 MPa\right)$$

where 560 sq. inch (0.361 m^2) is the metal area (cross section) of the inner shell in the HI-STAR 100 overpack.

The total axial stress, therefore, is

$$\sigma_{\mathrm{T}} = 917 + 492 = 1{,}409 \text{ psi} \qquad (\sigma_{\mathrm{T}} = 6.32 + 3.39 = 9.71 MPa)$$

Per Table 3.1.7, the allowable membrane stress intensity for a Level D event is 48,200 psi (332 MPa) at 400°F (204°C). Therefore, a factor of safety is calculated as

$$SF = \frac{48{,}200}{1{,}409} = 34.2 \qquad \left(SF = \frac{332}{9.71} = 34.2\right)$$

Sliding Analysis

An assessment of sliding of the HI-STAR 100 System on the ISFSI pad during a postulated limiting seismic event is performed using a one-dimensional "slider block on friction supported surface" model. The HI-STAR 100 is simulated as a rigid block of mass *m* placed on a surface which is subject to a sinusoidal acceleration of amplitude *a*. The apparent mass of the block is assumed to be reduced by a factor α to recognize the contribution of vertical acceleration in the most adverse manner (vertical acceleration acts to reduce the downward force on the friction interface). The equation of motion for such a "slider block" is given by

$$m\ddot{x} = R - m\,a\,\sin\omega t$$

where:

$\ddot{x}$: relative acceleration of the slider block (double dot denotes second derivative of displacement *x* in time)

a: amplitude of the sinusoidal acceleration input

ω: frequency of the seismic input motion (radians/sec)

t: time coordinate

R is the resistive Coulomb friction force which can reach a maximum value of μ (αmg) (μ = coefficient of friction) and which always acts in the direction of opposite to $\dot{x}(t)$.

Solution of the above equation can be obtained by standard numerical integration for specified values of m, a, ω and α. The following input values are used.

a = 0.354g

α = 0.875 = 1 - vertical acceleration - (vertical acceleration is 0.125g for net horizontal acceleration equal to 0.354 from the acceleration table provided in the foregoing)

m = 245,000 lbs/g (1,090 kN/g)

μ = 0.25

For establishing the appropriate value of ω, reference is made to the USAEC publication TID-7024, "Nuclear Reactor and Earthquakes", page 35, 1963, which states that the significant energy of all seismic events in the U.S. essentially lies in the range of 0.4 to 10 Hz. Taking the mid-point value

$$\omega = (2\pi)\,(0.5)\,(0.4+10) = 32.7 \text{ rad/sec.}$$

The numerical solution of the above equation yields the maximum displacement of the slider block x_{max} as 0.047 inches (1.19 mm), which is negligible compared to the spacing between casks.

Calculations performed at lower values of ω show an increase in x_{max} with reducing ω. At 1 Hz, for example, x_{max} = 1.236 inches (31.39 mm). It is apparent from the above that there is a large margin of safety against inter-module collision within the HI-STAR 100 arrays at an ISFSI, where the minimum installed spacing is approximately 4 feet (1.22 m) (Table 1.4.1).

3.4.8 Tornado Wind and Missile Impact (Load Case B in Table 3.1.1 and Load Case 06 in Table 3.1.5

During a tornado event, the HI-STAR 100 System is conservatively assumed to be subjected to a constant wind force. It is also subject to impacts by postulated missiles. The maximum wind speed is specified in Table 2.2.4 and the three missiles, designated as large, intermediate, and small, are described in Table 2.2.5.

The post impact response of the HI-STAR 100 System is required to assess stability.

Supplement 19 of [3.4.13] contains results for the post-impact response of the HI-STAR 100 where it is demonstrated there that the combination of tornado missile plus either steady tornado wind or instantaneous tornado pressure drop causes a rotation of the HI-STAR 100 to a maximum angle of inclination 18.23 degrees from vertical. This is less than the angle required to overturn the cask. The appropriate value for the drag coefficient used in the computation of the lateral force on the overpack from tornado wind is justified in Supplement 19 of [3.4.13].

Supplement 19 of [3.4.13] computes the maximum force acting on the projected area of the cask to be

F = 26,380 lbs (117.3 kN)

This is bounded by the seismic overturning force computed in Section 3.4.7. Therefore, the overpack stress analysis performed in Section 3.4.7 remains governing.

As discussed in Section 3.4.6 in relation to the flood event, the HI-STAR 100 overpack is inherently more stable when it is stored horizontally on an ISFSI pad. Therefore, the stability evaluation summarized above for a freestanding HI-STAR 100 overpack in the vertical orientation, under the combined effects of tornado wind and a large missile impact, is also bounding for a HI-STAR 100 overpack stored horizontally.

The penetration potential of the missile strikes (Load Case 06 in Table 3.1.5) is examined in Supplement 22 of [3.4.13]. It is shown in Supplement 22 of [3.4.13] that there will be no penetration of the intermediate shells surrounding the inner shell of the overpack or penetration of the top closure plate. Therefore, there will be no radiological release associated with any missile strikes during a tornado. The following results summarize the work in Supplement 22 of [3.4.13].

a. The small missile will dent any surface it impacts, but no significant puncture force is generated.

b. The following table summarizes the denting and penetration analysis performed for the intermediate missile in Supplement 22 of [3.4.13]. Denting is used to connote a local deformation mode encompassing material beyond the impacting missile envelope, while penetration is used to indicate a plug type penetration mechanism involving only the target material immediately under the impacting missile.

Intermediate Missile Strike – Denting and Penetration		
Location	Denting (in) (mm)	Penetration
Outer Enclosure Shell	2.77 (70.4)	Yes (> 0.5 in) (> 12.7 mm)
Intermediate shells	2.81 (71.4)	No (< 8.5 in) (< 216 mm)
Closure plate	3.00 (76.2)	No (< 6 in) (< 152 mm)

Since the intermediate missile generates a large puncture force for a short duration, the effect of this puncture force on the overpack closure bolts is examined in Supplement 21 of [3.4.13].

The primary stresses that arise due to an intermediate missile strike on the side of the overpack and in the center of the overpack top lid are also determined in Supplement 22 of [3.4.13]. It is demonstrated there that Level D stress limits are not exceeded in either the side shell or the top lid. The safety factor in the overpack inner shell, considered as a cantilever beam under tip load, is computed, as is the safety factor in the top lid, considered as a centrally loaded plate. The applied load, in each case, is the missile impact load. A summary of the results is given in the table below:

HI-STAR 100 Missile Impact - Global Stress Results (Load Case 06 in Table 3.1.5)			
Item	Value (ksi) (MPa)	Allowable (ksi) (MPa)	Safety Factor
Inner Shell - Side Strike	12.6 (86.9)	48.2 (332)	3.83
Intermediate Shell – Side Strike -	14.3 (98.6)	39.1 (270)	2.73
Closure Plate - (End Strike)	48.45 (334.1)	64.6 (445)	1.33

The above summary table does not include the circumferential fabrication stress since these have been designated as self-limiting, and therefore fall into the category of a secondary stress which need not be included in a Level D stress evaluation.

The calculated indentation and penetration results in the above tables are valid for both storage orientations. When the HI-STAR 100 overpack is stored horizontally, it is also vulnerable to a missile impact strike on the overpack bottom plate. However, the bottom plate is the same thickness, and is made from the same material, as the overpack closure plate, so the results for the closure plate can be extended to the bottom plate.

3.4.9 Non-Mechanistic Tip-over, Side and Vertical Drop Events

Pursuant to the provision in NUREG-1536, a non-mechanistic tip-over of a loaded HI-STAR 100 System on to the ISFSI pad is considered. Analyses are also performed to determine the maximum deceleration sustained by a side or vertical free fall of a loaded HI-STAR 100 System onto the ISFSI pad. The object of the analyses is to demonstrate that the plastic deformation in the fuel basket is sufficiently limited to permit the stored SNF to be retrieved by normal means and that there is no significant loss of radiation shielding in the system.

Ready retrievability of the fuel is presumed to be ensured if stress levels in the MPC structure remain below Level D limits during the postulated drop events.

Subsequent to the accident events, the overpack must be shown to contain the shielding so that unacceptable radiation levels do not result from the accident.

Appendix 3.A provides a description of the dynamic finite element analyses undertaken to establish the decelerations resulting from the postulated event. A non-mechanistic tip-over is considered together with a side and end drop of a loaded HI-STAR 100 System. A dynamic finite element analysis of each event is performed using a commercial finite element code well suited for such dynamic analyses with interface impact and non-linear material behavior. This code and methodology have been fully benchmarked against Lawrence Livermore Laboratories test data and correlation [3.4.12].

It is shown in Appendix 3.A that the peak deceleration is less than 60g's for tip-over. Table 3.A.3 shows that the maximum deceleration level at the top of the cask is 66.0 g's, while the corresponding deceleration level at the top of the fuel basket is 59.8 g's. For the case of a vertical drop from a height of 21" (533 mm), the bounding longitudinal deceleration is 52.3 g's. Finally, for a side drop from a height of 72” (1,829 mm), the maximum deceleration is 49.7 g’s.

Based on the above results, it is concluded that the design basis deceleration limit of 60g's (Table 3.1.2) provides a conservative input for Level D stress calculations to demonstrate retrievability of stored fuel.

3.4.10 Overpack Service Life

The term of the 10CFR72, Subpart L C of C, granted by the NRC is 20 years. Nonetheless, the HI-STAR 100 Overpack is designed for 40 years of service life, while satisfying the conservative design requirements defined in Chapter 2, including the regulatory requirements of 10CFR72. In addition, the overpack is designed, fabricated, and inspected under the comprehensive Quality Assurance Program discussed in Chapter 13 and in accordance with the applicable requirements of the ASME Codes. The pressure boundary (helium retention boundary) of the overpack is engineered to meet ASME Section III, Subsection NB (Class 1) stress intensity limits. Even though compliance to a less rigorous standard (such as the AISC Manual for Steel Construction) would be acceptable, all structural members of the HI-STAR 100 overpack located outside of the pressure boundary meet ASME Section III, Subsection NF stress limits. The aforementioned design and manufacturing measures assure high design margins, high quality fabrication, and verification of compliance through rigorous inspection and testing, as describe in Chapter 9. Technical Specifications defined in Chapter 12 assure that the integrity of the cask and the contained MPC are maintained throughout the components' service life.

The principal design considerations which bear on the adequacy of the overpack for the design basis service life are addressed as follows:

Exposure to Environmental Effects

All exposed surfaces of HI-STAR 100 are made from ferritic steels that are readily painted. Therefore, the potential of environmental vagaries are ruled out for HI-STAR 100. Under normal storage conditions, the bulk temperature of the HI-STAR 100 overpack will, because of its large thermal inertia, change very gradually with time. Therefore, material degradation from rapid thermal ramping conditions is not credible for the HI-STAR 100 overpack. The configuration of the overpack assures resistance to freeze-thaw degradation. In addition, the overpack is specifically designed for a full range of enveloping design basis natural phenomena which could occur over the 40-year service life of the overpack as defined in Section 2.2.3 and evaluated in Chapter 11.

Material Degradation

The relatively low neutron flux to which the overpack is subjected cannot produce measurable degradation of the cask's material properties and impair its intended safety function. Exposed carbon steel components are coated to prevent corrosion. The controlled environment of the ISFSI storage pad mitigates damage due to direct exposure to corrosive chemicals that may be present in other industrial applications.

Maintenance and Inspection Provisions

The requirements for periodic inspection and maintenance of the overpack throughout the 40-year service life are defined in Chapter 9. These requirements include provisions for routine inspection of the overpack exterior. ISFSIs located in areas subject to atmospheric conditions which may degrade the storage cask or canister should be evaluated by the licensee on a site-specific basis to determine the frequency for such inspections to assure long-term performance. In addition, the HI-STAR 100 System is designed for easy retrieval of the MPC from the overpack should it become necessary to perform more detailed inspections and repairs on the overpack.

The above findings are consistent with those of the NRC's Waste Confidence Decision Review [3.4.11], which concluded that dry storage systems designed, fabricated, inspected, and operate in accordance with such requirements are adequate for a 100-year service life while satisfying the requirements of 10CFR72.

3.4.11 MPC Service Life

The term of the 10CFR72, Subpart L C of C, granted by the NRC is 20 years. Nonetheless, the HI-STAR 100 MPC is designed for 40 years of service, while satisfying the conservative design requirements defined in Chapter 2, including the regulatory requirements of 10CFR72. Additional assurance of the integrity of the MPC and the contained SNF assemblies throughout the 40-year service life of the MPC is provided through the following:

- Design, fabrication, and inspection in accordance with the applicable requirements of the ASME Code as described in Chapter 2 assures high design margins.

- Fabrication and inspection performed in accordance with the comprehensive Quality Assurance program discussed in Chapter 13 assures competent compliance with the fabrication requirements.

- Use of materials with known characteristics, verified through rigorous inspection and testing, as described in Chapter 9, assures component compliance with design requirements.

- Use of welding procedures in full compliance with Sections III and IX of the ASME Code ensures high quality weld joints.

Technical Specifications, as defined in Chapter 12, have been developed and imposed on the MPC which assure that the integrity of the MPC and the contained SNF assemblies are maintained throughout the 40-year service life of the MPC.

The principal design considerations bearing on the adequacy of the MPC for the design basis service life are summarized below.

Corrosion

All MPC materials are fabricated from corrosion-resistant austenitic stainless steel and passivated aluminum. The corrosion-resistant characteristics of such materials for dry SNF storage canister applications, as well as the protection offered by these materials against other material degradation effects, are well established in the nuclear industry. The MPC is vacuum dried to remove all oxidizing liquids and gases and backfilled with dry inert helium at the time of closure to maintain an atmosphere in the MPC that provides corrosion protection for the SNF cladding throughout the dry storage period. The preservation of this non-corrosive atmosphere is assured by the inherent sealworthiness of the MPC confinement boundary integrity (there are no gasketed joints in the MPC).

Structural Fatigue

The passive non-cyclic nature of dry storage conditions does not subject the MPC to conditions that might lead to structural fatigue failure. Ambient temperature and insolation cycling during normal dry storage conditions and the resulting fluctuations in MPC thermal gradients and internal pressure is the only mechanism for fatigue. These low stress, high-cycle conditions cannot lead to a fatigue failure of the MPC which is made from stainless alloy stock (endurance limit well in excess of 20,000 psi (138 MPa)). All other off-normal or postulated accident conditions are infrequent or one-time occurrences which cannot produce fatigue failures. Finally, the MPC uses materials that are not susceptible to brittle fracture.

Maintenance of Helium Atmosphere

The inert helium atmosphere in the MPC provides a non-oxidizing environment for the SNF cladding to assure its integrity during long-term storage. The preservation of the helium atmosphere in the MPC is assured by the robust design of the MPC confinement boundary described in Section 7.1. Maintaining an inert environment in the MPC mitigates conditions that might otherwise lead to SNF cladding failures. The required mass quantity of helium backfilled into the canister at the time of closure as defined in the Technical Specification contained in Chapter 12, and the associated leak tightness requirements for the canister defined in the Technical Specification contained in Chapter 12, are specifically set down to assure that an inert helium atmosphere is maintained in the canister throughout a 40-year service life.

Allowable Fuel Cladding Temperatures

The helium atmosphere in the MPC promotes heat removal and thus reduces SNF cladding temperatures during dry storage. In addition, the SNF decay heat will substantially attenuate over a 40-year dry storage period. Maintaining the fuel cladding temperatures below allowable levels during long-term dry storage mitigates the damage mechanism that might otherwise lead to SNF cladding failures. The allowable long-term SNF cladding temperatures used for thermal acceptance of the MPC design are conservatively determined, as discussed in Section 4.3.

Neutron Absorber Boron Depletion

The effectiveness of the fixed borated neutron absorbing material used in the MPC fuel basket design requires that sufficient concentrations of boron be present to assure criticality safety during worst case design basis conditions over the 40-year service life of the MPC. Information on the characteristics of the borated neutron absorbing material used in the MPC fuel basket is provided in Section 1.2.1.3.1. The low neutron flux, which will continue to decay over time, to which this borated material is subjected, does not result in depletion of the material's available boron to prevent performing its intended safety function. In addition, the boron content of the material used in the criticality safety analysis is conservatively based on the minimum specified boron areal density (rather than the nominal), which is further reduced by 25% for analysis purposes, as described in Section 6.1. Analysis discussed in Section 6.3.2 demonstrates that the boron depletion in the neutron absorber material is negligible over a 50-year duration. Thus, sufficient levels of boron are present in the fuel basket neutron absorbing material to maintain criticality safety functions over the 40-year service life of the MPC.

The above findings are consistent with those of the NRC's Waste Confidence Decision Review, which concluded that dry storage systems designed, fabricated, inspected, and operated in the manner of the requirements set down in this document are adequate for a 100-year service life, while satisfying the requirements of 10CFR72.

Table 3.4.1

FINITE ELEMENTS IN THE MPC STRUCTURAL MODELS

MPC Type	Model Type		
Element Type	Basic	0 Degree Drop	45 Degree Drop
MPC-24	1068	1179	1178
BEAM3	1028	1028	1028
PLANE82	0	0	0
CONTAC12	40	38	38
CONTAC26	0	110	110
COMBIN14	0	3	2
MPC-32[1]	766	873	872
BEAM3	738	738	738
PLANE82	0	0	0
CONTAC12	28	27	24
CONTAC26	0	106	105
COMBIN14	0	2	5
MPC-68	1234	1347	1344
BEAM3	1174	1174	1174
PLANE82	16	16	16
CONTAC12	44	43	40
CONTAC26	0	112	111
COMBIN14	0	2	3

[1] The FE model for MPC-32 is adapted from Table 2.6.1 of [1.2.6].

Table 3.4.2

HI-STAR 100 SYSTEM MATERIAL COMPATIBILITY
WITH OPERATING ENVIRONMENTS

Material/Component	Fuel Pool (Borated and Unborated Water†)	ISFSI Pad (Open to Environment)
Alloy X: - MPC Fuel Basket - MPC Baseplate - MPC Shell - MPC Lid - MPC Fuel Spacers	Stainless steels have been extensively used in spent fuel storage pools with both borated and unborated water with no adverse reactions or interactions with spent fuel.	The MPC internal and external environment will be inert (helium) atmosphere. No adverse interactions identified.
Aluminum: - Conduction Elements	Aluminum and stainless steels form a galvanic couple. However, they are very close in the galvanic series chart and aluminum rapidly passivates in an aqueous environment forming a thin ceramic (Al_2O_3) barrier. Therefore, during the short time they are exposed to fuel pool water, significant corrosion or production of hydrogen is not expected (see operational requirements under "Boral" below).	In a non-aqueous atmosphere galvanic corrosion is not expected.
Boral: Metamic: - Neutron Absorber	The Boral will be passivated before installation in the fuel basket to minimize the amount of hydrogen released from the aluminum-water reaction to a non-combustible concentration during MPC lid welding or cutting operations. See Chapter 8 for additional requirements for combustible gas monitoring and required actions for control of combustible gas accumulation under the MPC lid. Stability of Metamic in the fuel pool environment has been demonstrated through extensive testing.	The Boral will be in a helium environment. No adverse reactions identified. No adverse reactions identified.

† HI-STAR 100 System short-term operating environment during loading and unloading.

Table 3.4.2 (continued)

HI-STAR 100 SYSTEM MATERIAL COMPATABILITY WITH OPERATING ENVIRONMENT

Material/Component	Fuel Pool (Borated and Unborated Water) †	ISFSI Pad (Open to Environment)
Steels: - SA350-LF3 - SA203-E - SA515 Grade 70 - SA516 Grade 70 - SA750 630 17-4 PH - SA564 630 17-4 PH - SA106 - SA193-B7 Overpack Body	All exposed steel surfaces (except seal areas, pocket trunnions (optional), and bolt locations) will be coated with paint specifically selected for performance in the operating environments. Even without coating, no adverse reactions (other than nominal corrosion) have been identified.	Internal surfaces of the overpack will be painted and maintained in an inert atmosphere. Exposed external surfaces (except those listed in fuel pool column) will be painted and will be maintained with a fully painted surface. No adverse reactions identified.

† HI-STAR 100 System short-term operating environment during loading and unloading.

Table 3.4.2 (continued)

HI-STAR 100 SYSTEM MATERIAL COMPATABILITY WITH OPERATING ENVIRONMENT

Material/Component	Fuel Pool (Borated and Unborated Water) †	ISFSI Pad (Open to Environment)
Stainless Steels: - SA240 304 - SA193 Grade B8 - 18-8 S/S Miscellaneous Components	Stainless steels have been extensively used in spent fuel storage pools with both borated and unborated water with no adverse reactions.	Stainless steel has a long proven history of corrosion resistance when exposed to the atmosphere. These materials are used for bolts and threaded inserts. No adverse reactions with steel have been identified. No impact on performance.
Nickel Alloy: - SB637-NO7718 Bolting	Bolts are not used in pool.	Exposed to weathering effects. No adverse reactions with overpack closure plate. No impact on performance.
Brass: - Rupture Disk	Small surface of rupture disk will be exposed. No significant adverse impact identified.	Exposed to external weathering. No loss of function expected. Disks inspected prior to transport.
Holtite-A: - Neutron Shield	The neutron shield is fully enclosed by the outer enclosure. No adverse reaction identified. No adverse reactions with thermal expansion foam or steel.	The neutron shield is fully enclosed in the outer enclosure. No adverse reaction identified. No adverse reactions with thermal expansion foam or steel.

† HI-STAR 100 System short-term operating environment during loading and unloading.

Table 3.4.2 (continued)

HI-STAR 100 SYSTEM MATERIAL COMPATABILITY WITH OPERATING ENVIRONMENT

Material/Component	Fuel Pool (Borated and Unborated Water) †	ISFSI Pad (Open to Environment)
Silicone Foam: - Thermal Expansion Foam	Fully enclosed in the outer enclosure. No adverse reaction identified. No adverse reactions with neutron shield or steel.	Foam is fully enclosed in outer enclosure. No adverse reaction identified. No adverse reactions with neutron shield or steel.
Paint: - Carboline 890 - Thermaline 450	Carboline 890 used for exterior surfaces. Acceptable performance for short-term exposure in mild borated pool water. Thermaline 450 selected for excellent high temperature resistance properties. Will only be exposed to demineralized water during in-pool operations as annulus is filled prior to placement in the spent fuel pool and the inflatable seal prevents fuel pool water in-leakage. No adverse interaction identified which could affect MPC/fuel assembly performance.	Good performance on exterior surfaces. Discoloration is not a concern. During storage, internal overpack surfaces will operate in an inert (helium) atmosphere. No adverse reaction identified.
Metallic Seals: - Alloy X750 - 304 S/S	Not installed or exposed during in-pool handling.	Seals enclosed by closure plate or port coverplates. Closure plate seals seat against stainless steel overlay surfaces. No degradation of seal integrity due to corrosion is expected.

† HI-STAR 100 System short-term operating environment during loading and unloading.

Table 3.4.3

FUEL BASKET RESULTS- GLOBAL MINIMUM OF SAFETY FACTORS

Load Case I.D.	**Loading**†	**Safety Factor**	**Location in FSAR where Results or Detailed Calculations are Presented**††
F1	T, T'	No Interference	3.4.4.2.1
F2	D + H	2.87	Table 3.4.9
F3			
F3.a	D + H' (end drop)	4.39	3.4.4.3.1.3
F3.b	D + H' (side drop 0^o)	1.19	Table 3.4.9
F3.c	D + H' (side drop 45^o)	1.29	Table 3.4.9

† The symbols used for the loading are defined in Table 2.2.13.
†† All Safety Factors for the Fuel Basket are conservatively evaluated using allowable stresses evaluated at 725 degrees F (385 degrees C).

Table 3.4.4

ENCLOSURE VESSEL RESULTS – GLOBAL MINIMUM OF SAFETY FACTORS

Load Case I.D.	Load Combination[†]	Safety Factor	Component ID and Location in FSAR where Results or Detailed Calculations are Presented[†††]	
E1				
E1.a	Design internal pressure, P_i	5.06[†††††] 1.33 1.36	Lid Baseplate Shell	Table 3.4.7 Supplement 15 of [3.4.14] Table 3.4.7
E1.b	Design external pressure, P_o	7.5[†††††] 1.2 1.2	Lid Baseplate Shell	Supplement 14 of [3.4.14] Table 3.4.7[††††] Table 3.4.15
E1.c	Design internal pressure, P_i , plus Temperature T	12.6 2.7 1.5	Lid Baseplate Shell	Table 3.4.8 Table 3.4.8 Table 3.4.8
E2	(P_i, P_o)[††] + D + H	1.8[†††††] 1.09 2.64 5.84	Lid Baseplate Shell Supports	Supplement 28 of [3.4.14] Supplement 15 of [3.4.14] Table 3.4.9 Table 3.4.9

† The symbols used for the loadings are defined in Table 2.2.13.

†† The notation (P_i, P_o) means that one or the other pressure is applied to determine the governing condition.

††† Safety Factors computed in Table 3.4.9 are conservatively based on the design temperature given in Table 3.1.17.

†††† Safety Factor obtained by multiplication of result for internal pressure by external pressure/internal pressure ratio.

††††† Minimum safety factor is based on dual lid configuration.

Table 3.4.4 (Continued)

ENCLOSURE VESSEL RESULTS – GLOBAL MINIMUM OF SAFETY FACTORS

Load Case I.D.	Load Combination†	Safety Factor	Component ID and Location in FSAR where Detailed Calculations are Presented††	
E3				
E3.a	$(P_i, P_o) + D + H'$, end drop	7.69†††	Lid	Table 3.4.13
		1.87	Baseplate	Table 3.4.12
		1.92	Shell	Table 3.4.15
E3.b	$(P_i, P_o) + D + H'$, side drop 0º	NA	Lid	
		NA	Baseplate	
		3.07	Shell	Table 3.4.9
		1.16	Supports	Table 3.4.9
E3.c	$(P_i, P_o) + D + H'$, side drop 45º	NA	Lid	
		NA	Baseplate	
		2.74	Shell	Table 3.4.9
		1.51	Supports	Table 3.4.9

† Symbols used in the loading are defined in Table 2.2.13.
†† Safety Factors computed in Table 3.4.9 conservatively use allowable stresses at the design temperature of 450 degrees F (232 degrees C) (Table 3.1.17)
††† Minimum safety factor is based on dual lid configuration.

Table 3.4.4 (Continued)

ENCLOSURE VESSEL RESULTS – GLOBAL MINIMUM OF SAFETY FACTORS

Load Case I.D.	**Load Combination**[†]	**Safety Factor**	**Component ID and Location in FSAR where Detailed Calculations are Presented**
E4	T	No restraint of free thermal expansion under normal heat or fire accident	3.4.4.2.1
E5	$(P_I^*, P_o^*) + D + T^*$	8.58[††] 1.17 1.18	Lid Table 3.4.13 Baseplate Table 3.4.12 Shell Table 3.4.15

† The symbols used for the loading are defined in Table 2.2.13.

†† Minimum safety factor is based on the dual lid configuration.

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Table 3.4.5

OVERPACK – GLOBAL MINIMUM SAFETY FACTORS

Load Case I.D.	Load Combination†	Safety Factor	Location in FSAR
01	(P_i, P_o)	2.86	Table 3.4.10
02	(P_i^*, P_o^*) + D + T*	5.50	Table 3.4.10
03	(P_i, P_o) + D + T + H	3.56	Table 3.4.6
04			
04.a	(P_i, P_o) + D + H (end drop)	1.27	Table 3.4.10
04.b	(P_i, P_o) + D + H (side drop)	1.48	Table 3.4.10
05	T	1.93	Table 3.4.6
06	M (small and medium penetrant missiles)	No effect on confinement boundary	Supplement 22 of [3.4.13]

† The symbols used for the loadings are defined in Table 2.2.13.

Table 3.4.6

OVERPACK SAFETY FACTORS TO INCORPORATE FABRICATION STRESS AND ACCIDENT TEMPERATURE[†]

Load Case	Inner Shell (Exterior Surface)			Inner Shell (Middle Surface)			Intermediate Shell		
	Value (ksi) (MPa)	Allowable (ksi) (MPa)	Safety Factor	Value (ksi) (MPa)	Allowable (ksi) (MPa)	Safety Factor	Value (ksi) (MPa)	Allowable (ksi) (MPa)	Safety Factor
01	19.1 (131.7)	68.7 (473.7)	3.60	12.19 (84.05)	68.7 (473.7)	5.64	12.74 (87.84)	52.5 (362.0)	4.12
03	19.28 (132.93)	68.7 (473.7)	3.56	12.30 (84.80)	68.7 (473.7)	5.59	14.61 (100.73)	52.5 (362.0)	3.59
05	35.68 (246.00)	68.7 (473.7)	1.93	26.65 (183.74)	68.7 (473.7)	2.58	NA	NA	NA

† The Value is obtained by adding the fabrication stress from Subsection 3.4.4.3.2.2 (absolute value) to the stress intensity from the finite element solution to compute a conservative modified stress intensity and then re-computing the safety factor based on allowable values for primary plus secondary stresses.

Table 3.4.7
STRESS INTENSITY RESULTS FOR CONFINEMENT BOUNDARY - INTERNAL PRESSURE ONLY (Load Case E1.a in Table 3.1.4)

Component Locations (Per Fig.3.4.44)	Calculated Value of Stress Intensity (psi) (MPa)	Category	Table 3.1.13 Allowable Value (psi)† (MPa)	Safety Factor (Allowable/Calculated)
Top Lid††				
A	3,282 (22.6)	$P_L + P_b$	30,000 (206.8)	9.14
Neutral Axis	40.4 (0.278)	P_m	20,000 (137.9)	495
B	3,210 (22.1)	$P_L + P_b$	30,000 (206.8)	9.34
C	1,374 (9.5)	$P_L + P_b$	30,000 (206.8)	21.8
Neutral Axis	1,462 (10.1)	P_m	20,000 (137.9)	13.6
D	5,920 (40.8)	$P_L + P_b$	30,000 (206.8)	5.06
Baseplate				
E	19,683 (135.7)	$P_L + P_b$	30,000 (206.8)	1.5
Neutral Axis	412 (2.8)	P_m	20,000 (137.9)	48.5
F	20,528 (141.5)	$P_L + P_b$	30,000 (206.8)	1.5
G	9,695 (66.8)	$P_L + P_b$	30,000 (206.8)	3.1
Neutral Axis	2,278 (15.7)	P_m	20,000 (137.9)	8.8
H	8,340 (57.5)	$P_L + P_b$	30,000 (206.8)	3.5

† Allowable stress intensity conservatively taken at 300 degrees F (149 degrees C).

†† The stresses in the top lid are reported for the dual lid configuration. The stresses for the single lid configuration are 50% less (see Subsection 3.4.4.3.1.2 for further details).

Table 3.4.7 (Continued)

STRESS INTENSITY RESULTS FOR CONFINEMENT BOUNDARY - INTERNAL PRESSURE ONLY (Load Case E1.a in Table 3.1.4)

Component Locations (Per Fig. 3.4.44)	Calculated Value of Stress Intensity (psi) (MPa)	Category	Table 3.1.13 Allowable Value (psi)† (MPa)	Safety Factor (Allowable/Calculated)
Canister				
I	6,860 (47.3)	P_m	18,700 (128.9)	2.72
Upper Bending Boundary Layer Region	7,189 (49.6) 7,044 (48.6)	$P_L + P_b + Q$ $P_L + P_b$	30,000 (206.8) 20,000 (137.9)	4.2 2.8
Lower Bending Boundary Layer Region	43,986 (303.3) 10,621 (73.2)	$P_L + P_b + Q$ $P_L + P_b$	60,000 (413.7) 30,000 (206.8)	1.36 2.82

† Allowable stress intensity conservatively based at 300 degrees F (149 degrees C) except for Location I where allowable stress intensity values are based on 400 degree F (204 degrees C).

Table 3.4.8

PRIMARY AND SECONDARY STRESS INTENSITY RESULTS FOR CONFINEMENT BOUNDARY - PRESSURE PLUS THERMAL LOADING (Load Case E1.c in Table 3.1.4)

Locations (Per Fig. 3.4.44)	Calculated Value of Stress Intensity (psi) (MPa)	Category	Allowable Stress Intensity (psi)† (MPa)	Safety Factor (Allowable/Calculated)
Top Lid††				
A	4,750 (32.8)	$P_L + P_b + Q$	60,000 (413.7)	12.6
Neutral Axis	1,530 (10.5)	P_L	30,000 (206.8)	19.6
B	2,140 (14.8)	$P_L + P_b + Q$	60,000 (413.7)	28.0
C	2,200 (15.2)	$P_L + P_b + Q$	60,000 (413.7)	27.2
Neutral Axis	3,650 (25.2)	P_L	30,000 (206.8)	8.21
D	7,034 (48.5)	$P_L + P_b + Q$	60,000 (413.7)	8.52
Baseplate				
E	21,921 (151.1)	$P_L + P_b + Q$	60,000 (413.7)	2.7
Neutral Axis	1,287 (8.9)	P_L	30,000 (206.8)	23.3
F	19,386 (133.7)	$P_L + P_b + Q$	60,000 (413.7)	3.1
G	6,152 (42.4)	$P_m + P_L$	60,000 (413.7)	9.8
Neutral Axis	4,564 (31.5)	P_L	30,000 (206.8)	6.6
H	11,306 (78.0)	$P_L + P_b + Q$	60,000 (413.7)	5.3

† Allowable stresses based on temperature of 300 degrees F (149 degrees C).

†† The stresses in the top lid are reported for the dual lid configuration. The stresses for the single lid configuration are 50% less (see Subsection 3.4.4.3.1.2 for further details).

Table 3.4.8 (Continued)

PRIMARY AND SECONDARY STRESS INTENSITY RESULTS FOR
CONFINEMENT BOUNDARY - PRESSURE PLUS THERMAL LOADING (Load Case E1.c in Table 3.1.4)

Locations (Per Fig. 3.4.44)	Calculated Value of Stress Intensity (psi) (MPa)	Category	Allowable Stress Intensity (psi)† (MPa)	Safety Factor (Allowable/Calculated)
<u>Canister</u>				
I	6,900 (48)	P_L	28,100 (194)	4.07
Upper Bending Boundary Layer Region	6,490 (45) 4,834 (33)	$P_L + P_b + Q$ P_L	60,000 (414) 30,000 (207)	9.2 6.2
Lower Bending Boundary Layer Region	39,929 (275) 7,480 (52)	$P_L + P_b + Q$ P_L	60,000 (414) 30,000 (207)	1.5 4.0

†Allowable stresses based on temperature of 300 degrees F (149 degrees C) except at Location I where the temperatures are based on 400 degrees F (204 degrees C).

Table 3.4.9

FINITE ELEMENT ANALYSIS RESULTS
MINIMUM SAFETY FACTORS FOR MPC COMPONENTS (Load Cases from Tables 3.1.3 and 3.1.4)

Component – Stress Result	MPC-24		
	Handling Load Load Cases F2 or E2	0 Degree Side Drop Load Cases F3.b or E3.b	45 Degree Side Drop Load Cases F3.c or E3.c
Fuel Basket – Primary Membrane (P_m)	45.5(396)	2.80(1134)	3.85(396)
Fuel Basket – Local Membrane Plus Primary Bending ($P_L + P_b$)	11.9(75)	1.19(1065)	1.29(433)
Enclosure Vessel - Primary Membrane (P_m)	2.67(1370)	6.43(1277)	6.88(1370)
Enclosure Vessel – Local Membrane Plus Primary Bending ($P_L + P_b$)	3.56(1343)	4.24(1276)	4.28(1295)
Basket Supports – Primary Membrane (P_m)	N/A	N/A	N/A
Basket Supports – Local Membrane Plus Primary Bending ($P_L + P_b$)	N/A	N/A	N/A

Notes:
1. Corresponding ANSYS element number shown in parentheses.

Table 3.4.9 (continued)

FINITE ELEMENT ANALYSIS RESULTS
MINIMUM SAFETY FACTORS FOR MPC COMPONENTS (Load Cases from Tables 3.1.3 and 3.1.4)

Component - Stress Result	MPC-68		
	Handling Load Load Cases F2 or E2	0 Degree Side Drop Load Cases F3.b or E3.b	45 Degree Side Drop Load Cases F3.c or E3.c
Fuel Basket - Primary Membrane (P_m)	40.0 (798)	3.07 (1603)	4.30 (1603)
Fuel Basket - Local Membrane Plus Primary Bending ($P_L + P_b$)	2.87 (438)	2.64 (1033)	1.56 (774)
Enclosure Vessel - Primary Membrane (P_m)	2.64 (1747)	5.64 (1770)	7.12 (1864)
Enclosure Vessel - Local Membrane Plus Primary Bending ($P_L + P_b$)	2.96 (1864)	3.07 (1770)	2.74 (1866)
Basket Supports - Primary Membrane (P_m)	5.84 (1714)	6.67 (1699)	8.67 (1644)
Basket Supports - Local Membrane Plus Primary Bending ($P_L + P_b$)	9.01 (1713)	1.16 (1704)	1.51 (1649)

Notes:

1. Corresponding ANSYS element number shown in parentheses.

Table 3.4.9 (continued)

FINITE ELEMENT ANALYSIS RESULTS
MINIMUM SAFETY FACTORS FOR MPC COMPONENTS (Load Cases from Tables 3.1.3 and 3.1.4)

Component - Stress Result	MPC-32		
	Handling Load Load Cases F2 or E2	0 Degree Side Drop Load Cases F3.b or E3.b	45 Degree Side Drop Load Cases F3.c or E3.c
Fuel Basket - Primary Membrane (P_m)	39.7 (715)	2.78 (715)	3.90 (691)
Fuel Basket - Local Membrane Plus Primary Bending ($P_L + P_b$)	3.38 (294)	1.19 (678)	1.28 (235)
Enclosure Vessel - Primary Membrane (P_m)	2.63 (951)	5.77 (938)	6.95 (1039)
Enclosure Vessel - Local Membrane Plus Primary Bending ($P_L + P_b$)	3.00 (979)	2.14 (1021)	3.56 (973)
Basket Supports - Primary Membrane (P_m)	5.83 (836)	2.72 (908)	3.83 (878)
Basket Supports - Local Membrane Plus Primary Bending ($P_L + P_b$)	4.58 (878)	3.89 (908)	4.75 (888)

Notes:

1. Corresponding ANSYS element number shown in parentheses.

Table 3.4.10

FINITE ELEMENT RESULTS
MINIMUM SAFETY FACTORS FOR OVERPACK COMPONENTS UNDER VARIOUS LOADS (Load Case from Table 3.1.5)

Component - Stress Result	Internal Pressure – Load Case 01	Accident Internal Pressure – Load Case 02	Accident External Pressure - Load Case 02	Handling Load – Load Case 03	End Drop Load Case 04.a	Side Drop – Load Case 04.b	Thermal Load – Load Case 05
Lid – Local Membrane Plus Primary Bending ($P_L + P_b$)	2.86 (501)	5.50 (501)	14.3 (501)	4.45 (501)	9.2 (501)	3.68 (501)	2.55 (479)
Inner Shell - Local Membrane Plus Primary Bending ($P_L + P_b$)	12.1 (11023)	24.6 (11023)	13.6 (47)	11.3 (8338)	2.20 (10790)	1.70 (47)	3.72 (10925)
Inner Shell - Primary Membrane (P_m)	13.7 (281)	27.7 (281)	17.4 (10791)	12.7 (2969)	2.37 (48)	2.65 (2969)	2.98 (11024)
Intermediate Shells – Local Membrane Plus Primary Bending ($P_L + P_b$)	17.2 (11025)	36.6 (11025)	24.3 (49)	7.76 (285)	3.47 (10792)	1.48 (51)	2.42 (10796)
Baseplate - Local Membrane Plus Primary Bending ($P_L + P_b$)	10.6 (1)	17.9 (1)	7.77 (1)	18.8 (1)	1.27 (1)	4.98 (1)	2.07 (27)
Enclosure Shell - Primary Membrane (P_m)	35.1 (288)	72.7 (288)	41.4 (55)	13.2 (288)	8.47 (55)	1.55 (288)	2.03 (5428)

Notes:

1. Corresponding ANSYS node number shown in parentheses.

Table 3.4.11

SAFETY FACTORS FROM MISCELLANEOUS CALCULATIONS

Item	Loading	Safety Factor	FSAR or Calc Package Location Where Details are Provided
MPC Closure Ring	Internal Pressure	1.41	Supplement 14 of [3.4.14]
Fuel Basket Panels	Elastic Stability	3.59	3.4.4.3.1.3
MPC Top Lid Weld	Lifting	2.29	Supplement 14 of [3.4.14]
Fuel Support Spacers	Compression	1.14	Supplement 16 of [3.4.14]
MPC Cover Plates in MPC Lid	Accident Condition Internal Pressure	1.35	Supplement 40 of [3.4.14]
MPC Top Closure	10CFR71 Top End Drop (Transport) (Provided for Information Only)	2.8/(1.4)†	Supplement 14 of [3.4.14]
MPC Cover Plate Weld	Accident Condition Internal Pressure	2.52	Supplement 40 of [3.4.14]

† Results are presented for both the single and dual lid configurations (in parentheses). Refer to Subsection 3.4.4.3.1.2 for further information.

Table 3.4.12

MPC BASEPLATE MINIMUM SAFETY FACTORS FOR LOAD CASES E3 AND E5

Item	Value (ksi) (MPa)	Allowable (ksi) (MPa)	Safety Factor
Center of Baseplate – Primary Bending (Load Case E3)	35.93 (247.73)	67.32 (464.16)	1.87
Center of Baseplate – Primary Bending (Load Case E5)	46.32 (319.36)	54.23 (373.90)	1.17

Details of calculation are in Subsection 3.4.4.3.1.4.

Table 3.4.13

MPC TOP CLOSURE LID MINIMUM SAFETY FACTORS FOR LOAD CASES E3 AND E5

MPC Top Closure Lid – Minimum Safety Factors – Load Cases E3, E5			
Item	Stress (ksi) (MPa) or Load (lb.) (kN)	Allowable Stress (ksi) (MPa) or Load Capacity (lb.) (kN)	Safety Factor
Lid Bending Stress – Load Case E3.a	3.35/(7.94) (23.10/(54.74))	61.05 (420.92)	18.2/(7.69)
Lid Peripheral Weld Load – Load Case E3.a	624,000 (2,276)	1,477,000†† (6,570)	2.37
Lid Bending Stress – Load Case E5	3.158/(6.316) (21.77/(43.55))	54.225 (373.87)	17.17/(8.58)
Lid-to-Lid Peripheral Weld Load – Load Case E3.a	312,000 (1,388)	443,200††† (1,971)	1.42
Closure Ring Bending Stress – Load Case E1.a†	20.0 (137.9)	28.1 (193.7)	1.41
Closure Ring Weld Load – Load Case E1.a	140,956 (627)	316,400 (1,407)	2.24

Details of calculation are in Subsection 3.4.4.3.1.5

† The closure ring is only subject to load subsequent to a postulated loss of integrity in the "NB" pressure boundary (such as a leak in the MPC lid that is joined to the shell using a volumetrically examined groove weld). Nevertheless, the stress results are compared to Level A allowables for conservatism. The pressure loading is assumed to correspond to the Design Pressure, which as stated before, bounds both normal and off-normal conditions of storage.

†† Based on 0.625" (15.88 mm) single groove weld and conservatively includes a quality factor of 0.45.

††† This is a non-Code weld; limit is based on a 0.1875 inch (4.76 mm) groove weld and includes a quality factor of 0.45 for additional conservatism.

Table 3.4.14

MPC FUEL SPACERS – MINIMUM SAFETY FACTORS FOR LOAD CASES F2 AND F3.a

Fuel Spacers – Minimum Safety Factors (Load Cases F2 and F3.a)			
Item	Load (lb.) (kN)	Capacity (lb.) (kN)	Safety Factor
Axial Load – Level A	16,800 (75)	19,250 (86)	1.14
Elastic Stability – Level D – Lower Spacer	100,800 (448)	563,000 (2,504)	5.58
Elastic Stability – Level D – Upper Spacer	100,800 (448)	577,000 (2,567)	5.72

Details of calculation are in Subsection 3.4.4.3.1.6

Table 3.4.15

MPC SHELL STABILITY SAFETY FACTORS FROM ASME CODE CASE N-284

MPC Shell – Elastic/Plastic Stability (ASME Code Case N-284) – Minimum Safety Factors			
Item	Value	Allowable†	Safety Factor
Load Case E3.a (Yield)	0.698	1.34	1.92
Load Case E5 (Stability Interaction Equation)	0.845	1.0	1.18
Load Case E1.b (Stability Interaction Curve)	0.832	1.0	1.20

† We note that for Load Case E3.a, the yield strength criteria in the Code Case N-284 method governs. In this event, we include the safety factor 1.34, built into the Code Case, in the tabular result in order to obtain the actual safety factor with respect to the yield strength of the material.

Details of calculation are in Subsection 3.4.4.3.1.7

Table 3.4.16

OVERPACK FABRICATION STRESS – MINIMUM SAFETY FACTORS

Fabrication Stresses in Overpack Shells –Minimum Safety Factors (Level A Service Condition at Assembly Temperature)			
Item	Value (ksi) (MPa)	Allowable (ksi) (MPa) - (Note 3)	Safety Factor
First Intermediate Shell (Note 1)	11.22 (77.36)	52.5 (362.0)	4.68
Fourth Intermediate Shell (Note 1)	7.79 (53.71)	52.5 (362.0)	6.74
Inner Shell Mid Plane (Note 2)	10.6 (73.1)	69.9 (481.9)	6.59
Inner Shell Outer Surface (Note 2)	16.27 (112.18)	69.9 (481.9)	4.30

Notes:

1. The fabrication stress is a tensile circumferential stress.
2. The fabrication stress is a compressive circumferential stress
3. Fabrication stresses are self-limiting and are therefore classified as “secondary” and are compared to 3 times the membrane stress or stress intensity.

Details of calculation are in Subsection 3.4.4.3.2.2

Table 3.4.17

OVERPACK CLOSURE BOLT MINIMUM SAFETY FACTORS

Overpack Closure Bolt - Minimum Safety Factors	
Combined Load Case	Safety Factor on Bolt Tension
Normal (Load Case 01 in Table 3.1.5)	1.44
Top Closure Puncture (Load Case 06 in Table 3.1.5)	1.86
Drop (Load Case 04.a in Table 3.1.5)	1.30

Details of calculations are in Subsection 3.4.4.3.2.3

Table 3.4.18

OVERPACK CLOSURE PLATE – SAFETY FACTOR FOR LOAD CASE 04.a

Bending Stress in Overpack Closure Plate – Closed Form Solution (Load Case 04.a)			
Item	Value (ksi) (MPa)	Allowable (ksi) (MPa)	Safety Factor
Stress at Center of Plate	5.25 (36.2)	70.0 (482.6)	13.3

Details of calculations are in Subsection 3.4.4.3.2.4

Table 3.4.19

OVERPACK INNER SHELL SAFETY FACTORS FROM ASME CODE CASE N-284

†

Code Case N-284 Minimum Safety Factors – (Load Cases 02, 03 and 04.a in Table 3.1.5)			
Item	Calculated Interaction Value	Allowable Interaction Value†	Safety Factor against Yield†
Load Case 02 in Table 3.1.5	0.577	1.34	2.32
Load Case 03 in Table 3.1.5	0.613	2.0	3.26
Load Case 04 in Table 3.1.5	0.62	1.34	2.21

† We note that in computing the safety factor against yield for this table, we have included the safety factor implicit in the Code Case N-284 allowable interaction equation. We note also that the safety factors given above from the Code Case analysis are all safety factors against the circumferential or longitudinal stresses reaching the material yield stress. The actual safety factors against instability are larger than the factors reported in the table as can be seen by a perusal of Supplement 23 of [3.4.13]. Finally, we note that fabrication stresses have been included in the stability calculations even though these stresses are self-limiting. Therefore, all results corresponding to the calculated stability interaction equations in Supplement 23 of [3.4.13] are conservatively high.

Details of calculations are in Subsection 3.4.4.3.2.5

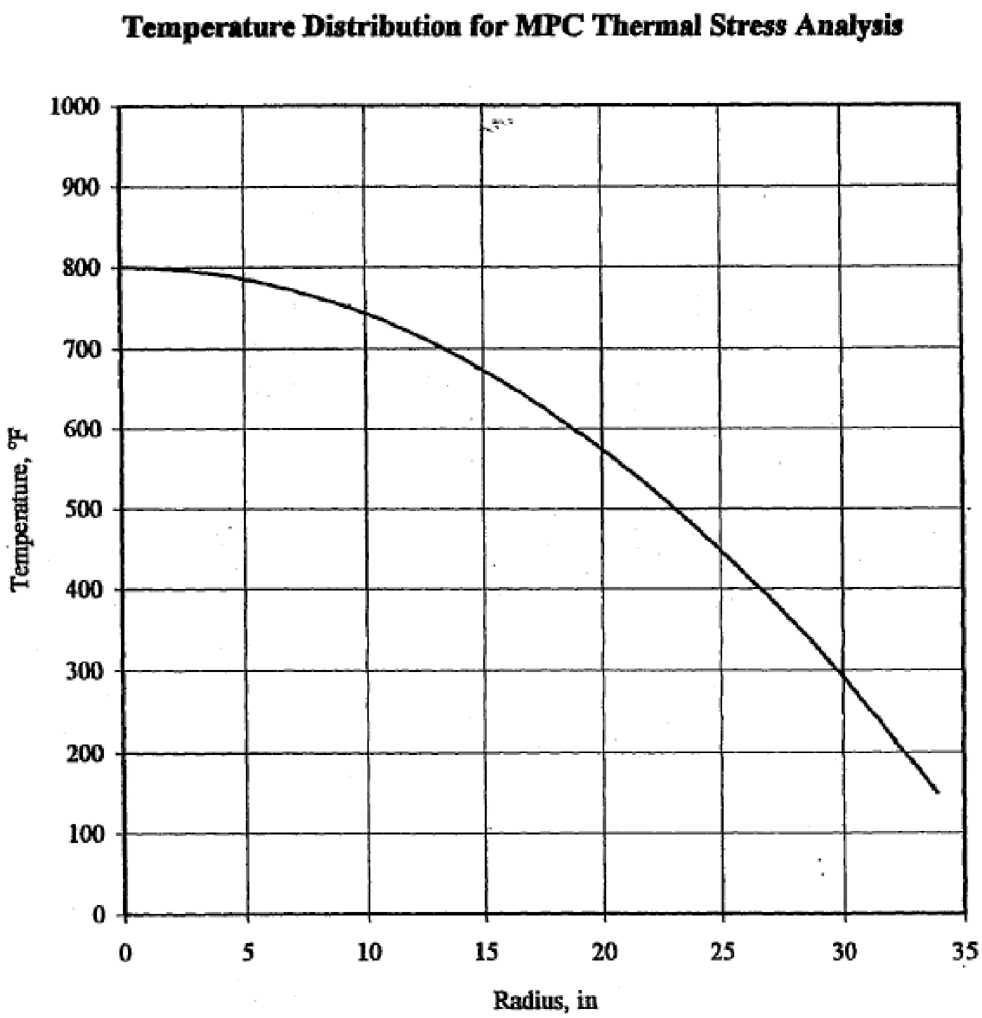

FIGURE 3.4.1; TEMPERATURE DISTRIBUTION FOR MPC THERMAL STRESS ANALYSIS

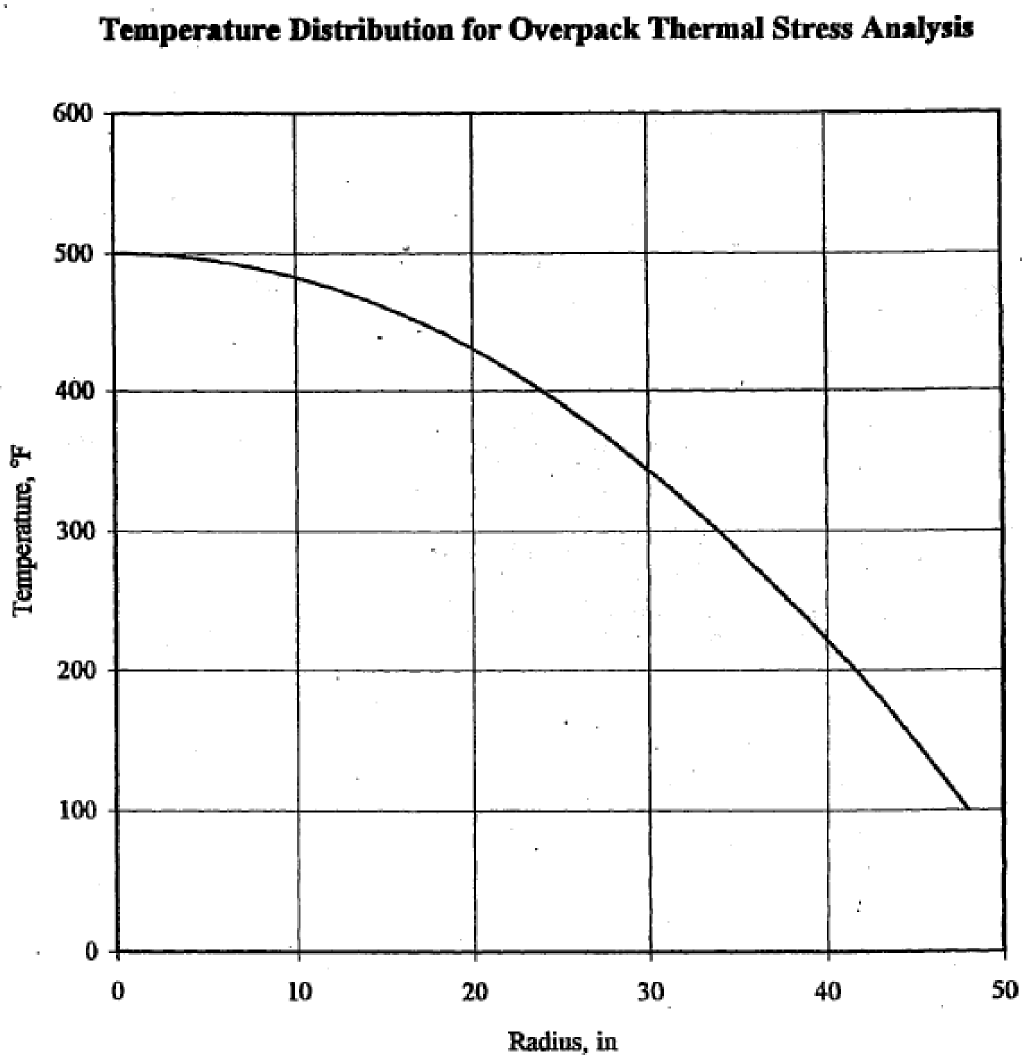

FIGURE 3.4.2; TEMPERATURE DISTRIBUTION FOR OVERPACK THERMAL STRESS ANALYSIS

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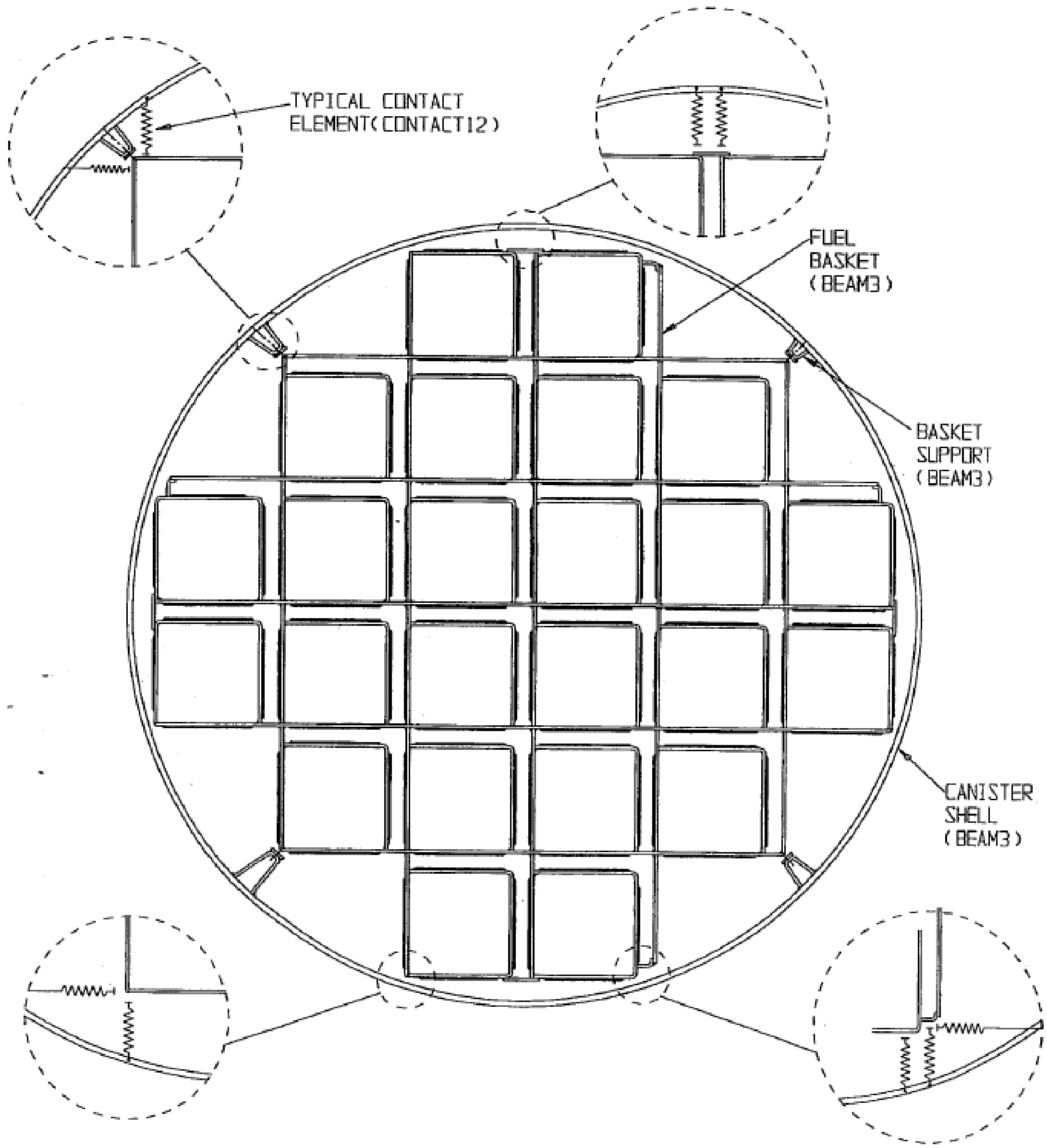

FIGURE 3.4.3: FINITE ELEMENT MODEL OF MPC-24

(BASIC MODEL)

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FIGURE 3.4.4;

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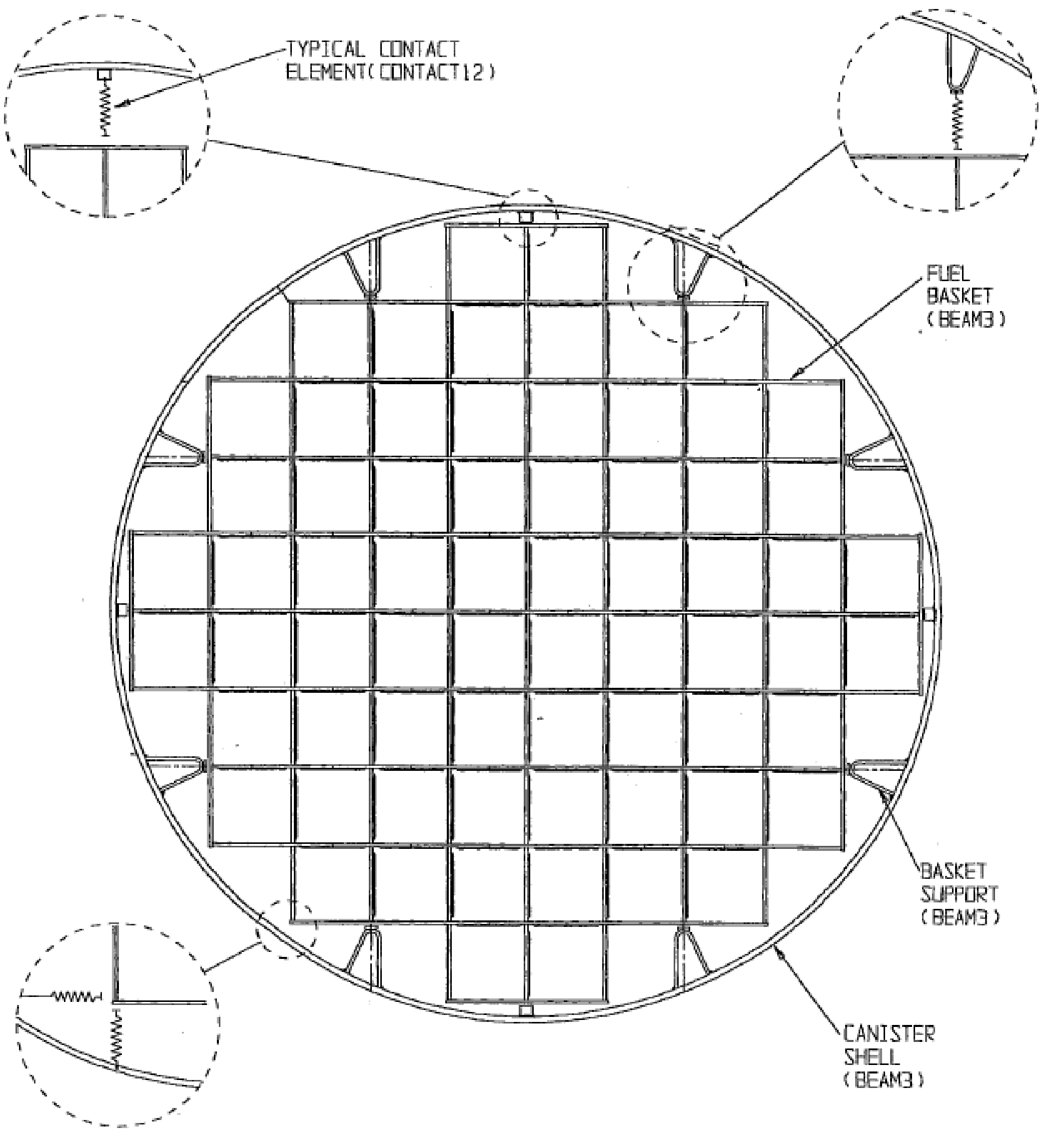

FIGURE 3.4.5: FINITE ELEMENT MODEL OF MPC-68

(BASIC MODEL)

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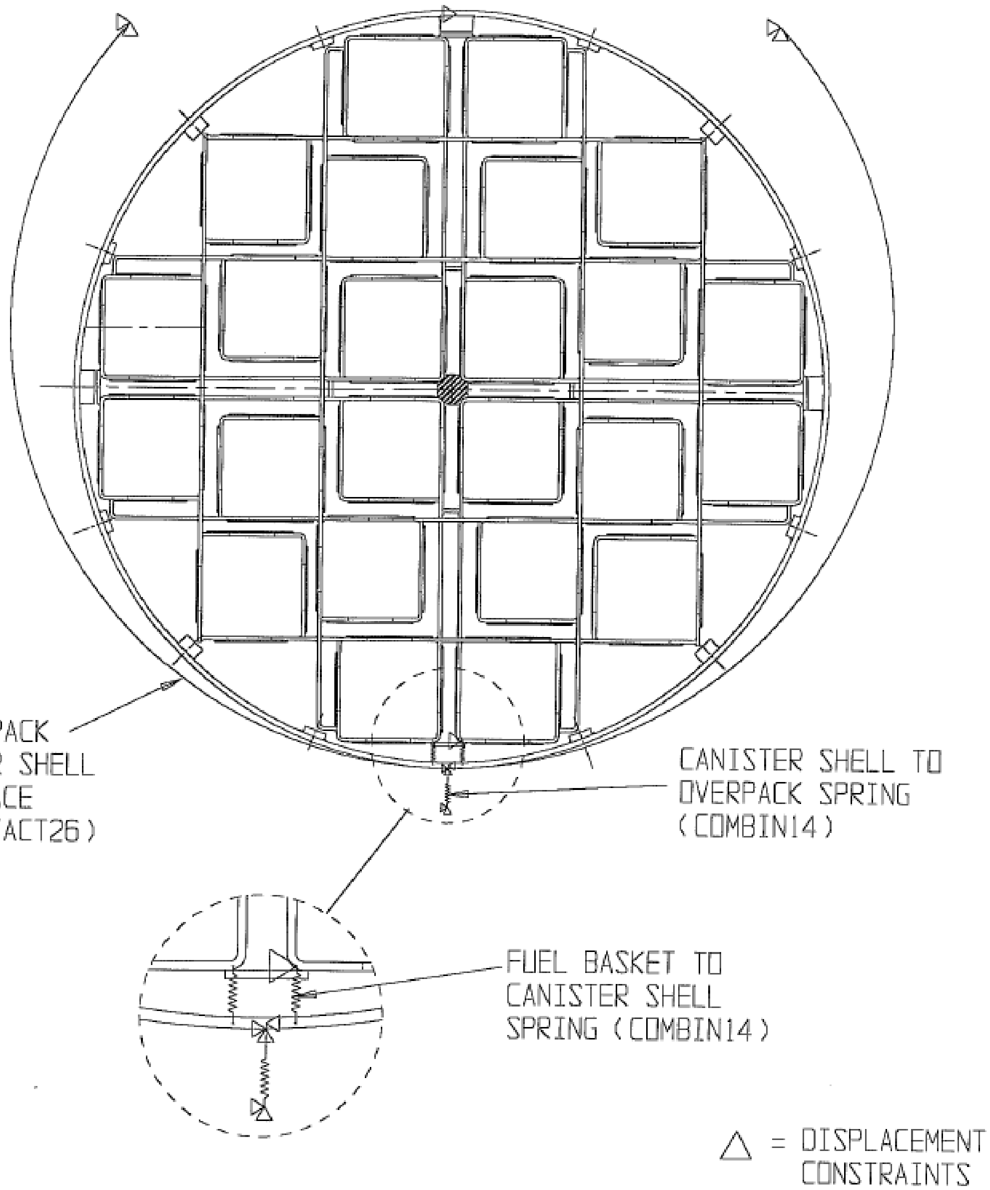

FIGURE 3.4.6: FINITE ELEMENT MODEL OF MPC-24

(0 DEGREE DROP MODEL)

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FIGURE 3.4.7;

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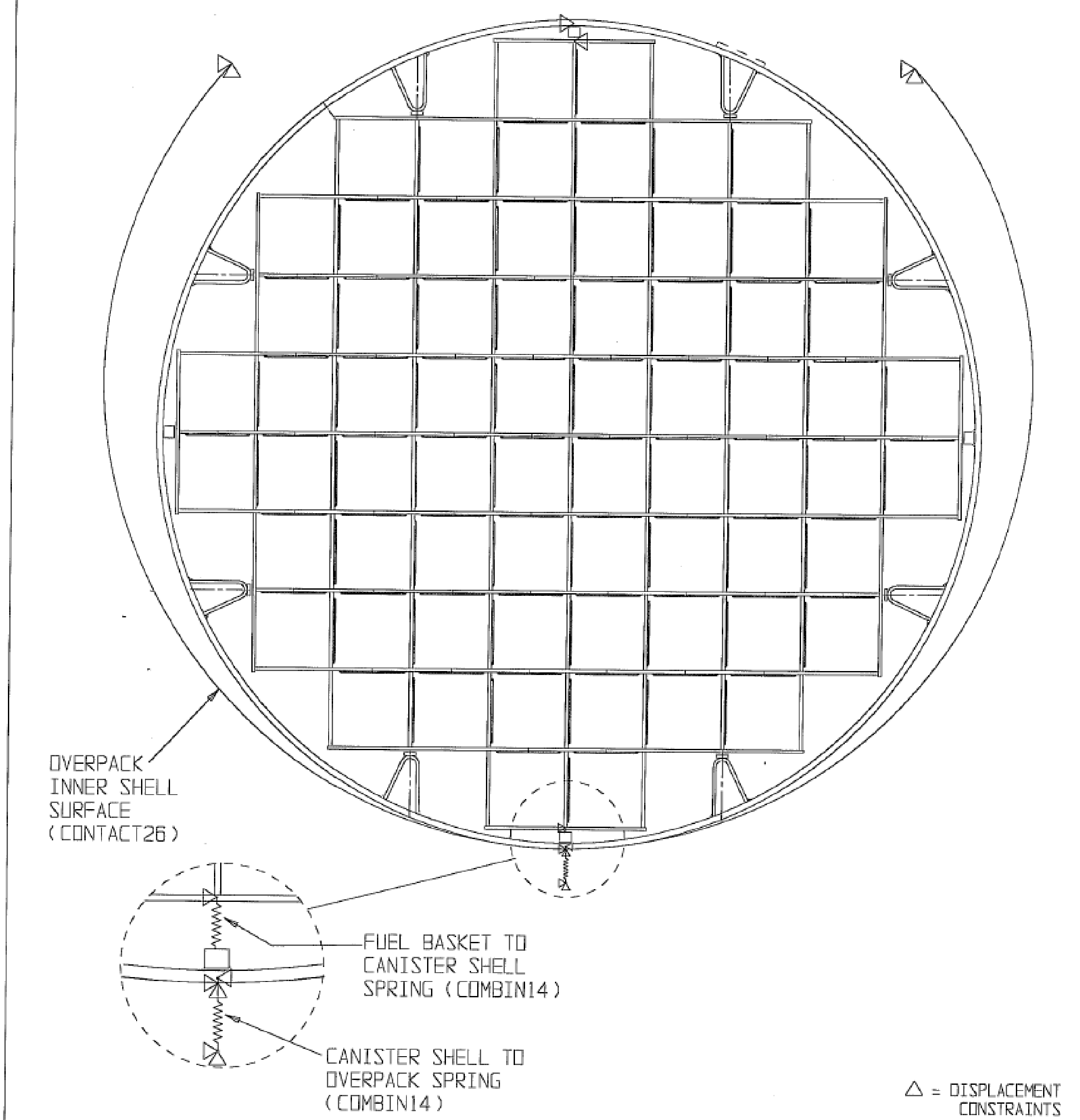

FIGURE 3.4.8; FINITE ELEMENT MODEL OF MPC-68

(0 DEGREE DROP MODEL)

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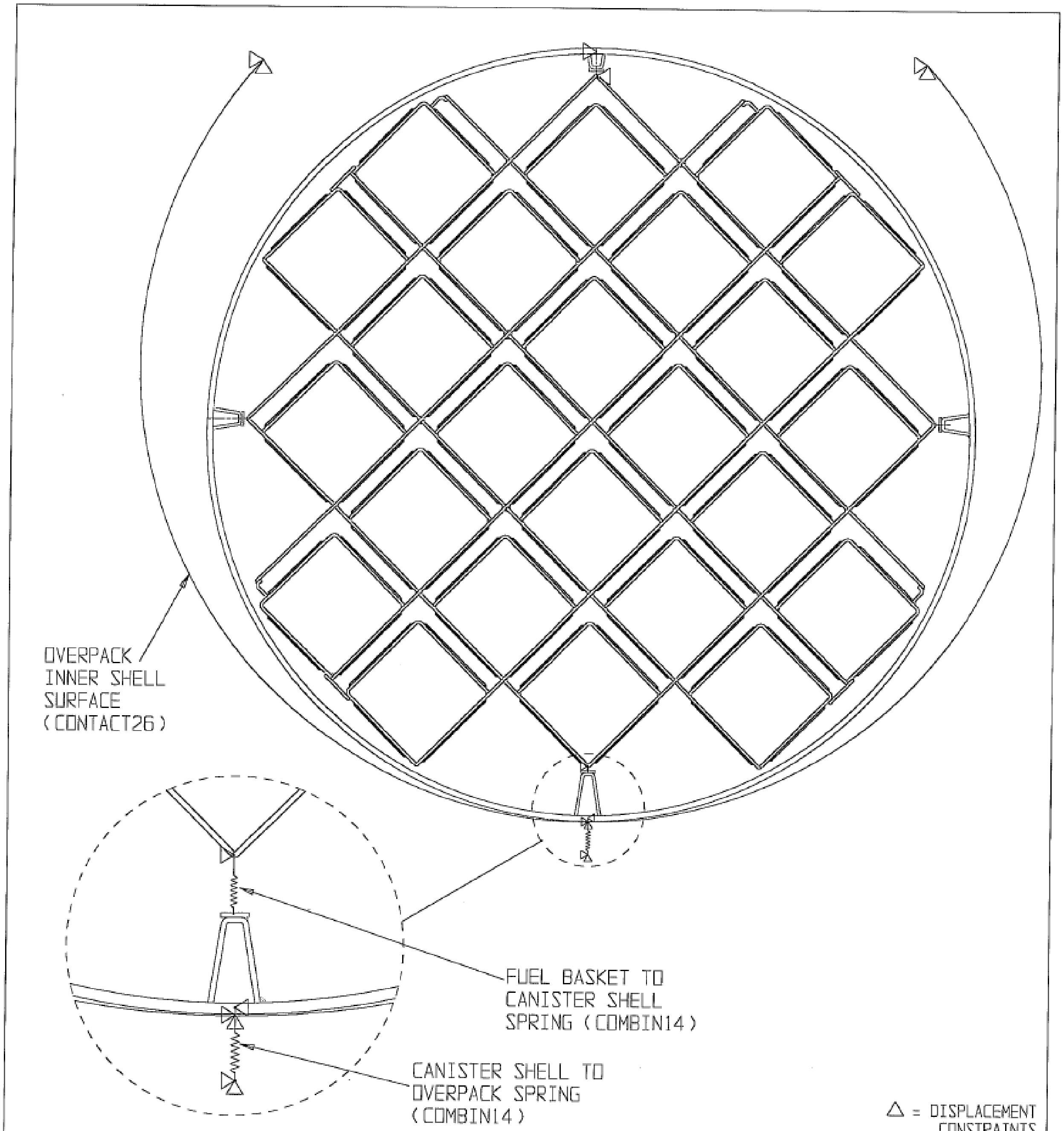

FIGURE 3.4.9; FINITE ELEMENT MODEL OF MPC-24

(45 DEGREE DROP MODEL)

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FIGURE 3.4.10; DELETED

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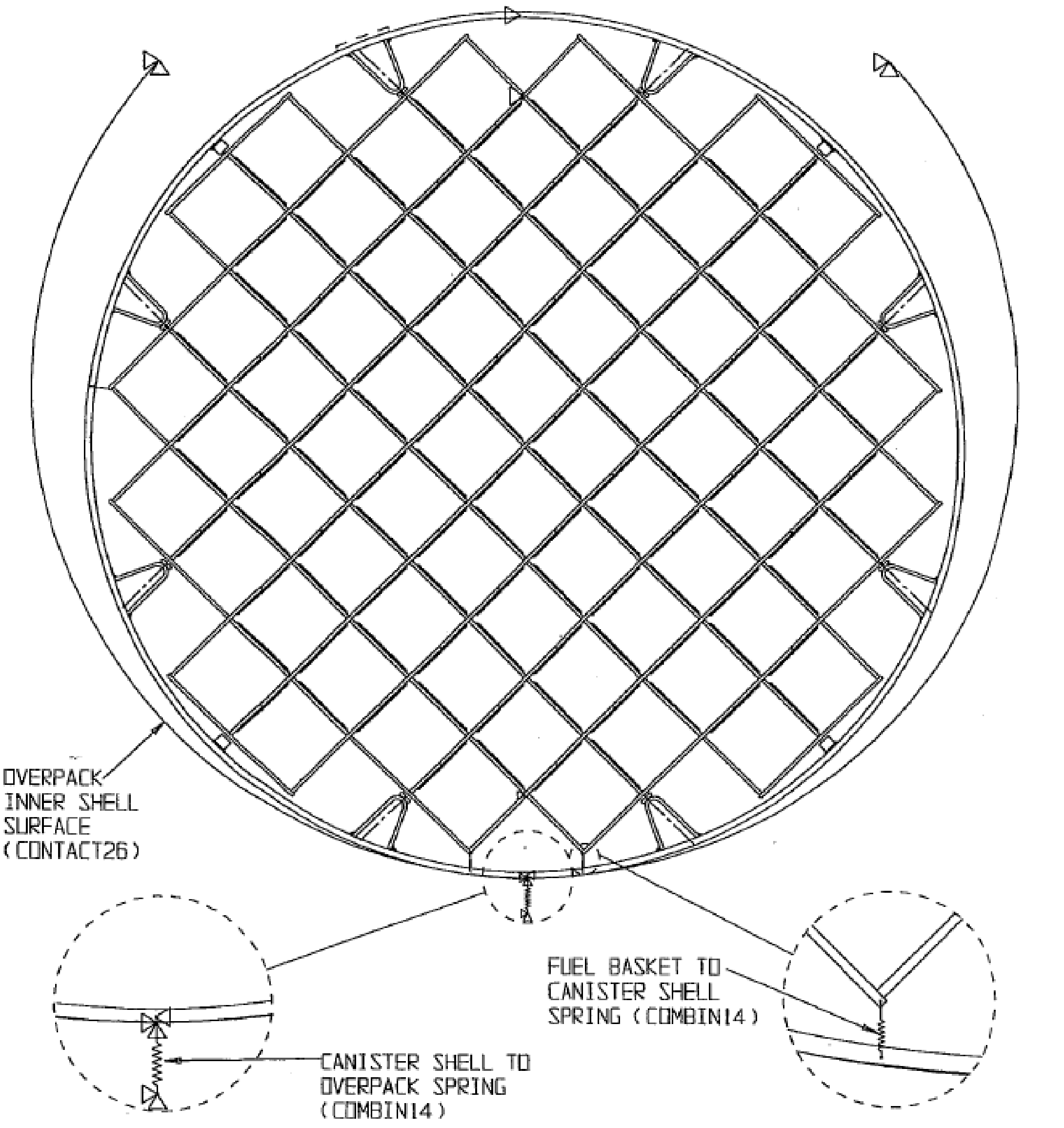

FIGURE 3.4.11: FINITE ELEMENT MODEL OF MPC-68

(45 DEGREE DROP MODEL)

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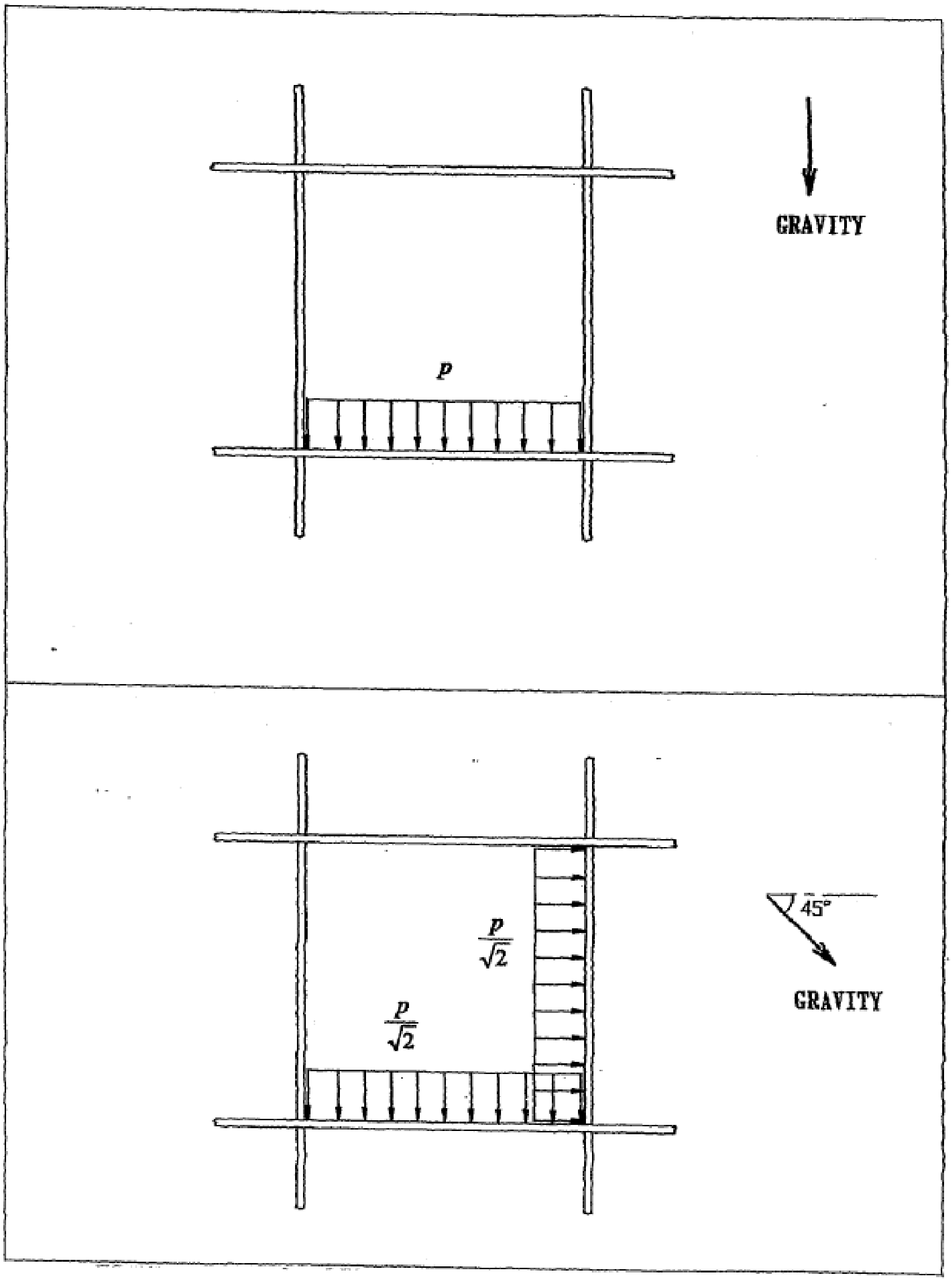

FIGURE 3.4.12: DETAIL OF FUEL ASSEMBLY PRESSURE LOAD ON MPC BASKET

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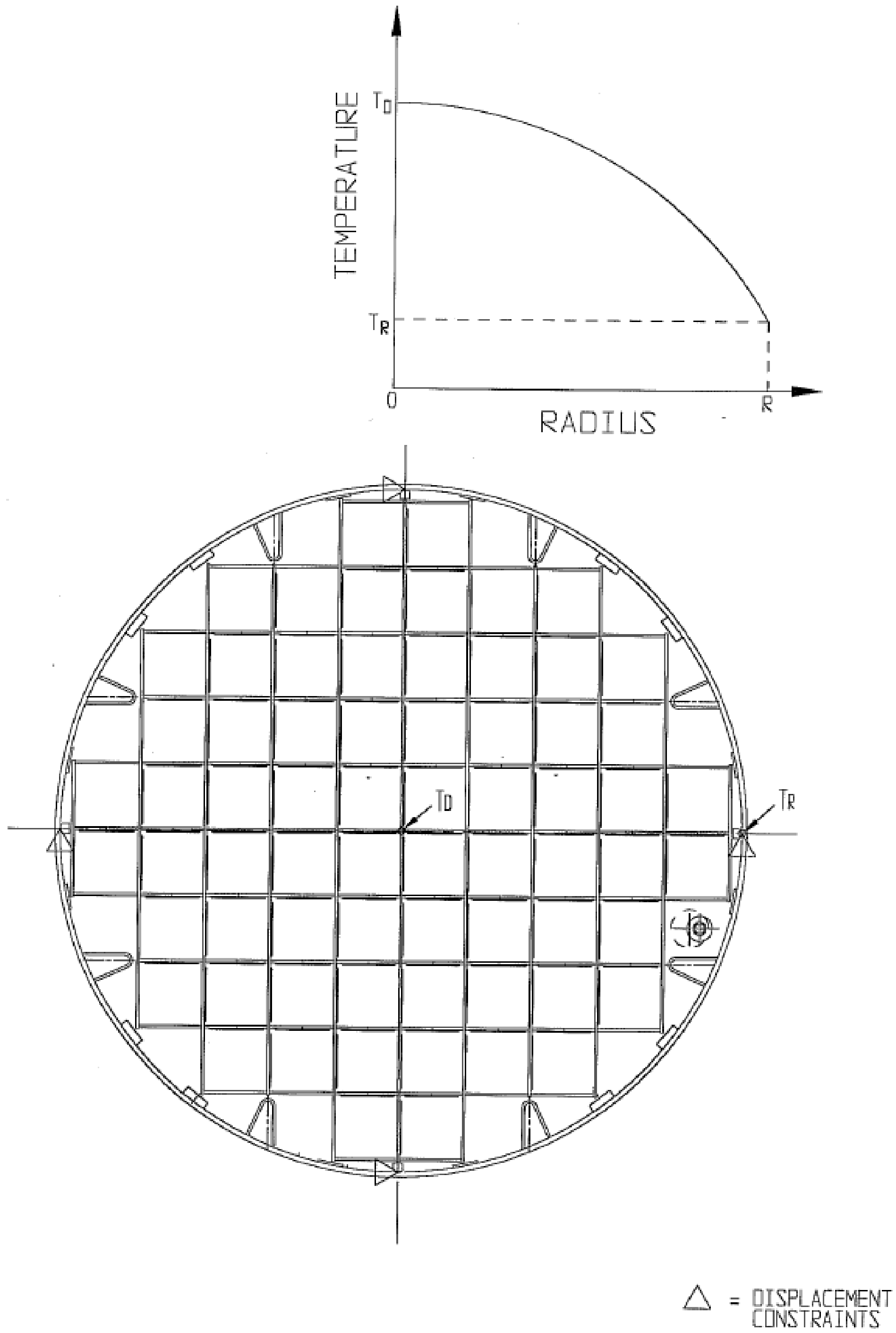

FIGURE 3.4.13; MPC THERMAL LOAD

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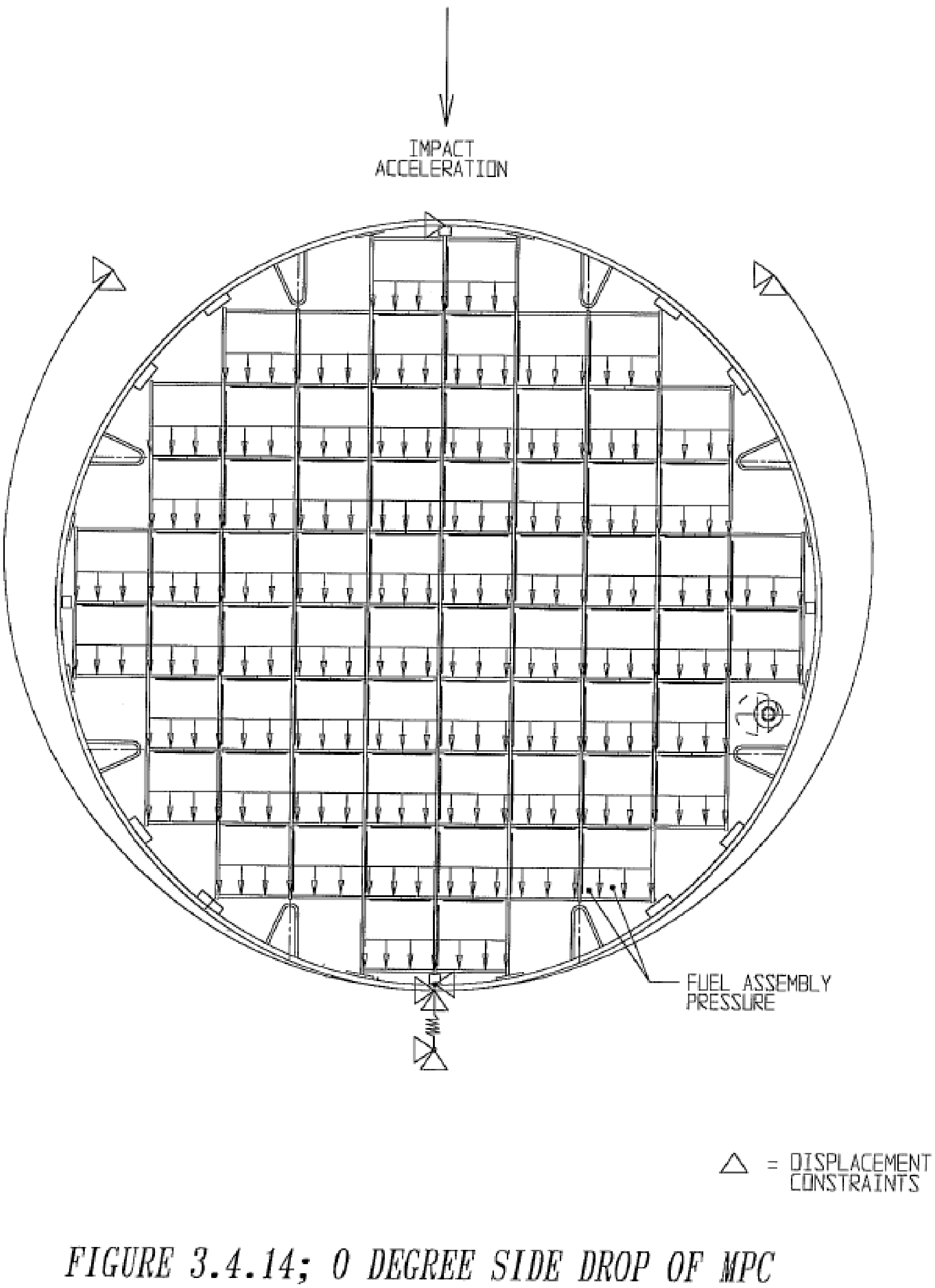

FIGURE 3.4.14; 0 DEGREE SIDE DROP OF MPC

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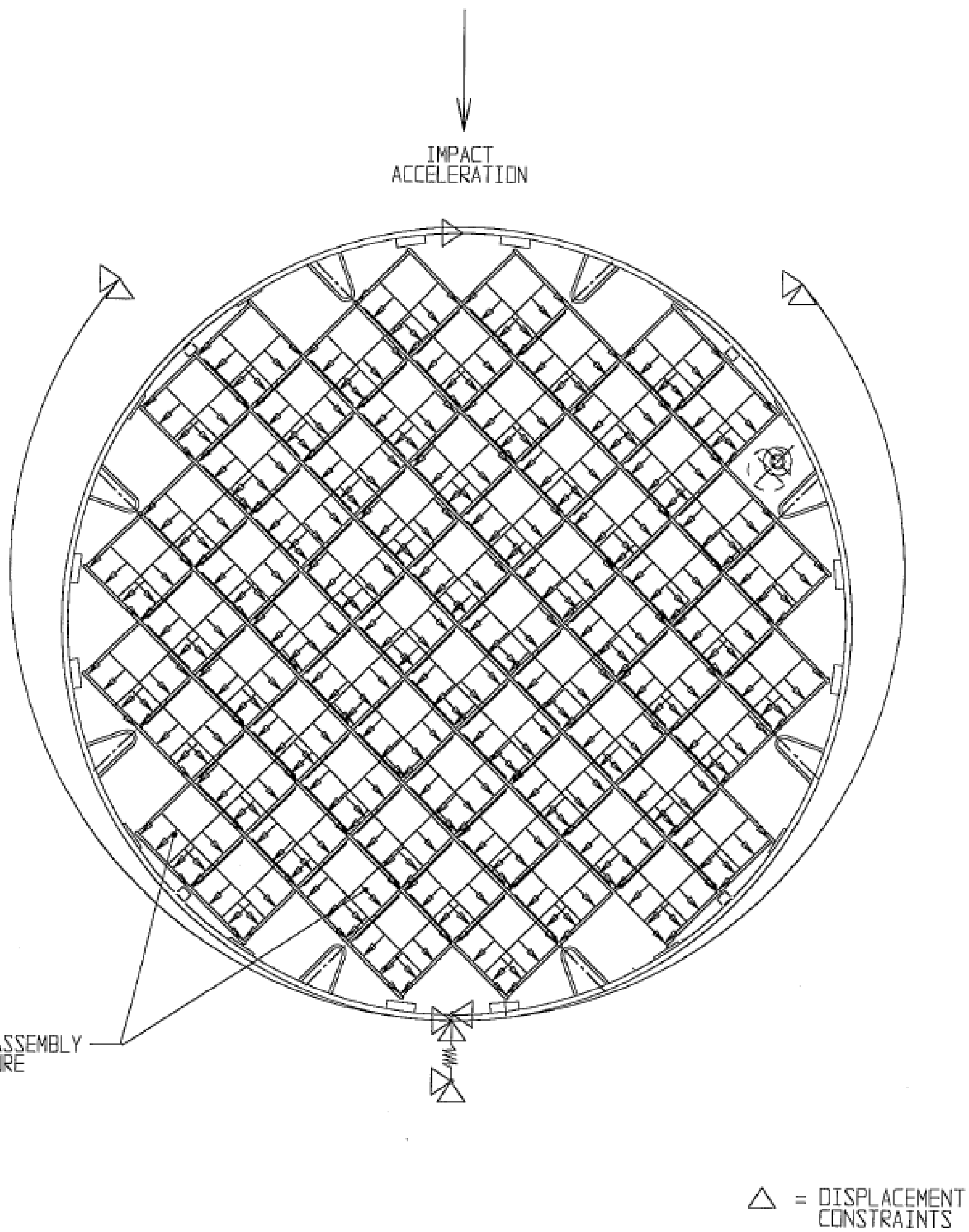

FIGURE 3.4.15; 45 DEGREE SIDE DROP OF MPC

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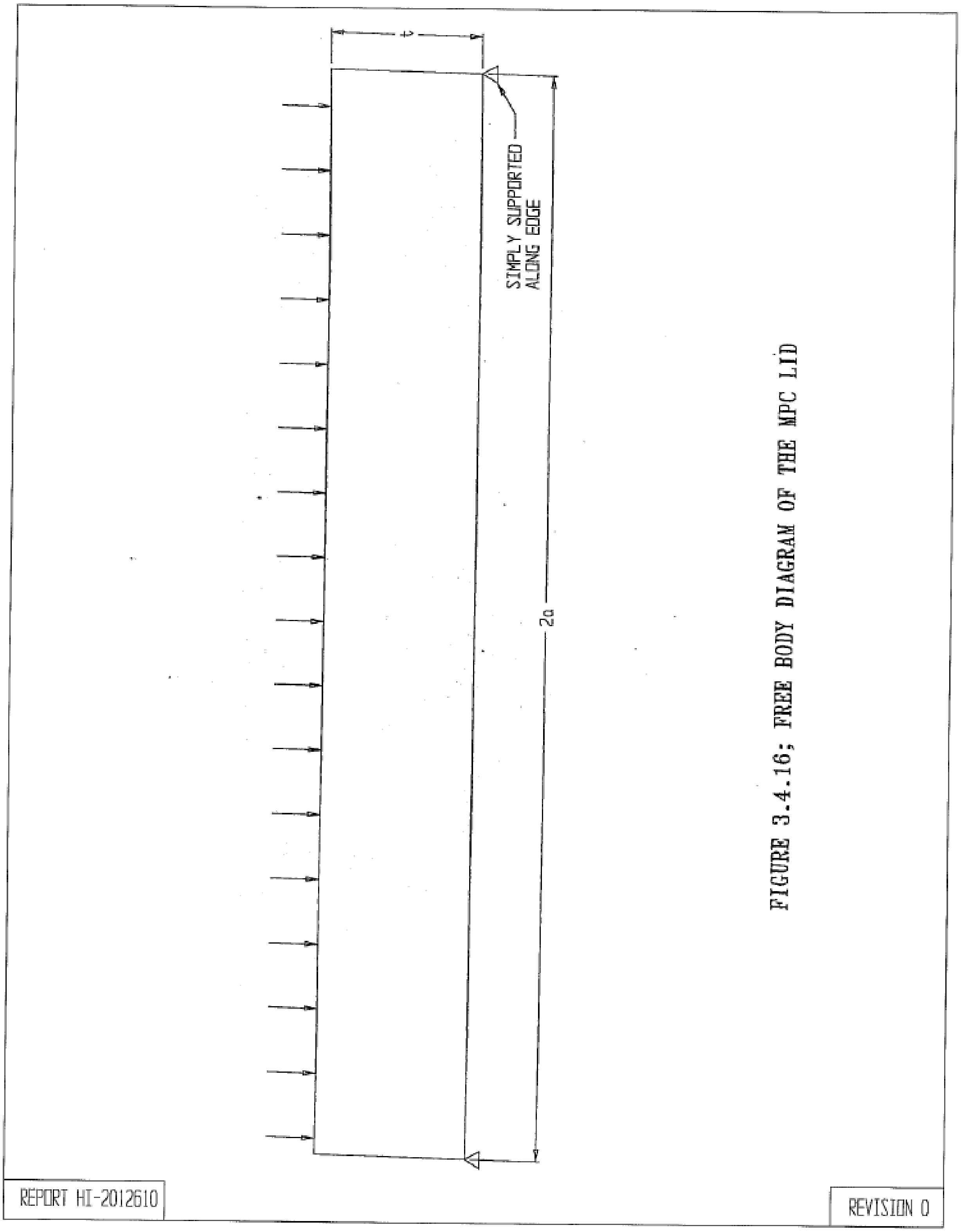

FIGURE 3.4.16; FREE BODY DIAGRAM OF THE MPC LID

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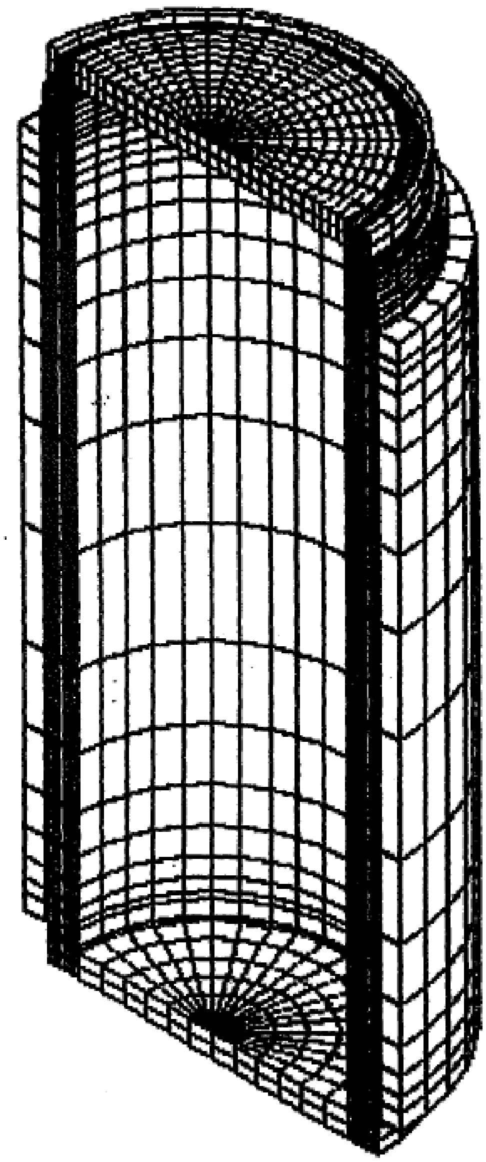

FIGURE 3.4.17; OVERPACK FINITE ELEMENT MODEL

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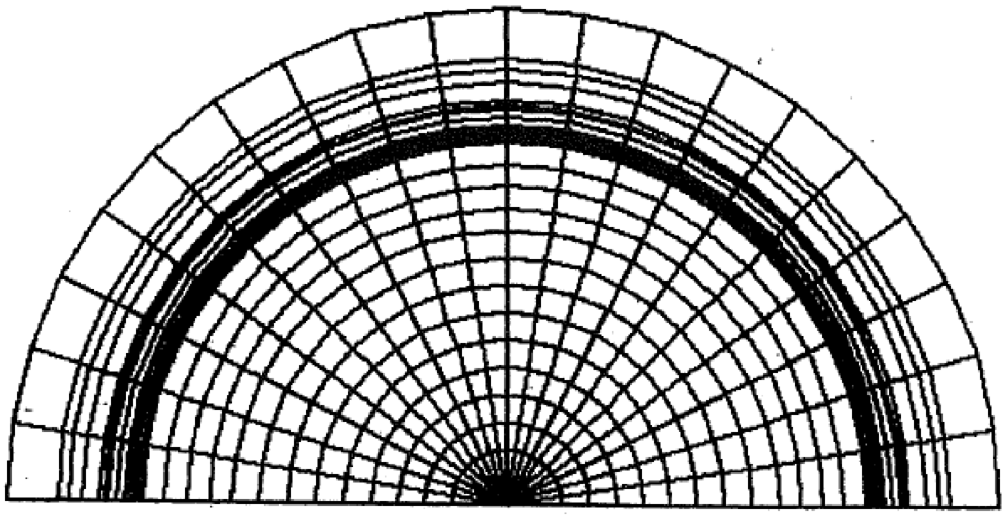

FIGURE 3.4.18; OVERPACK FINITE ELEMENT MODEL

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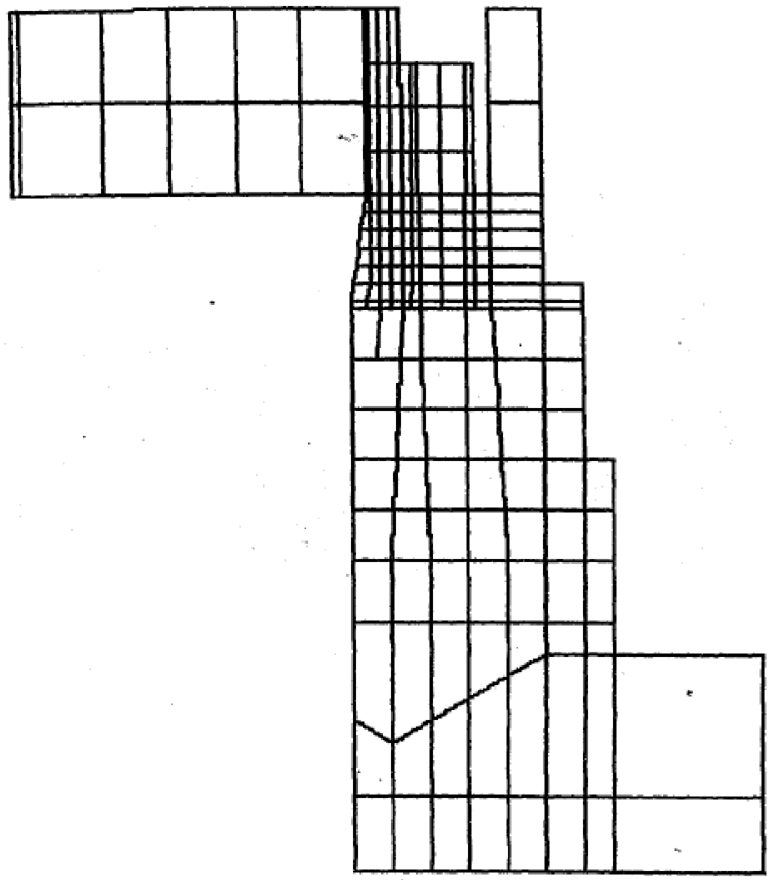

FIGURE 3.4.19; OVERPACK FINITE ELEMENT MODEL

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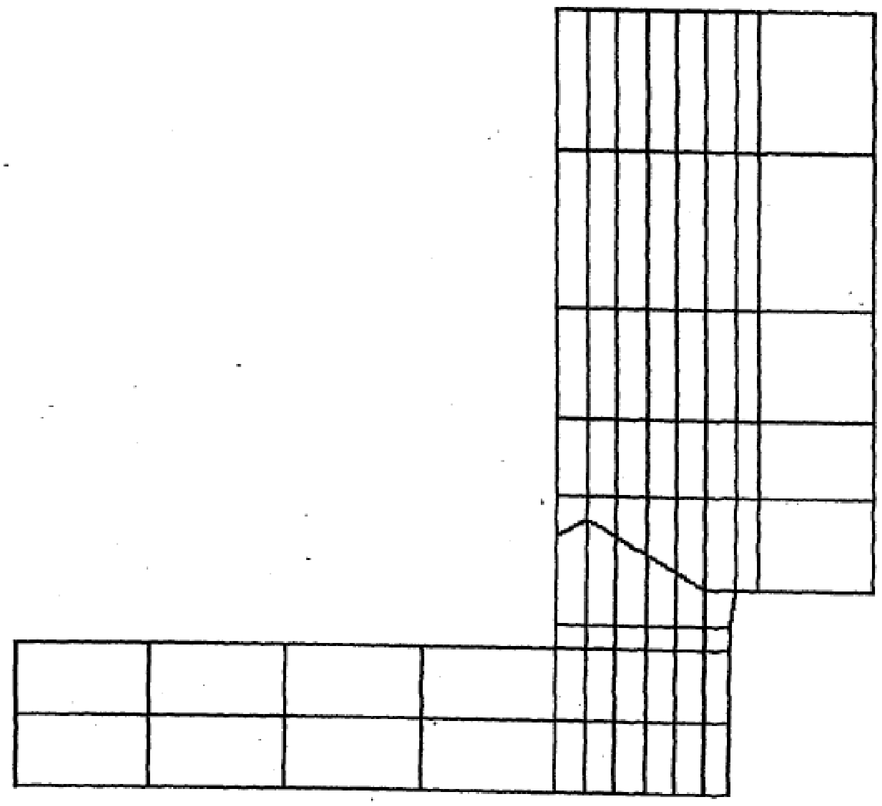

FIGURE 3.4.20; OVERPACK FINITE ELEMENT MODEL

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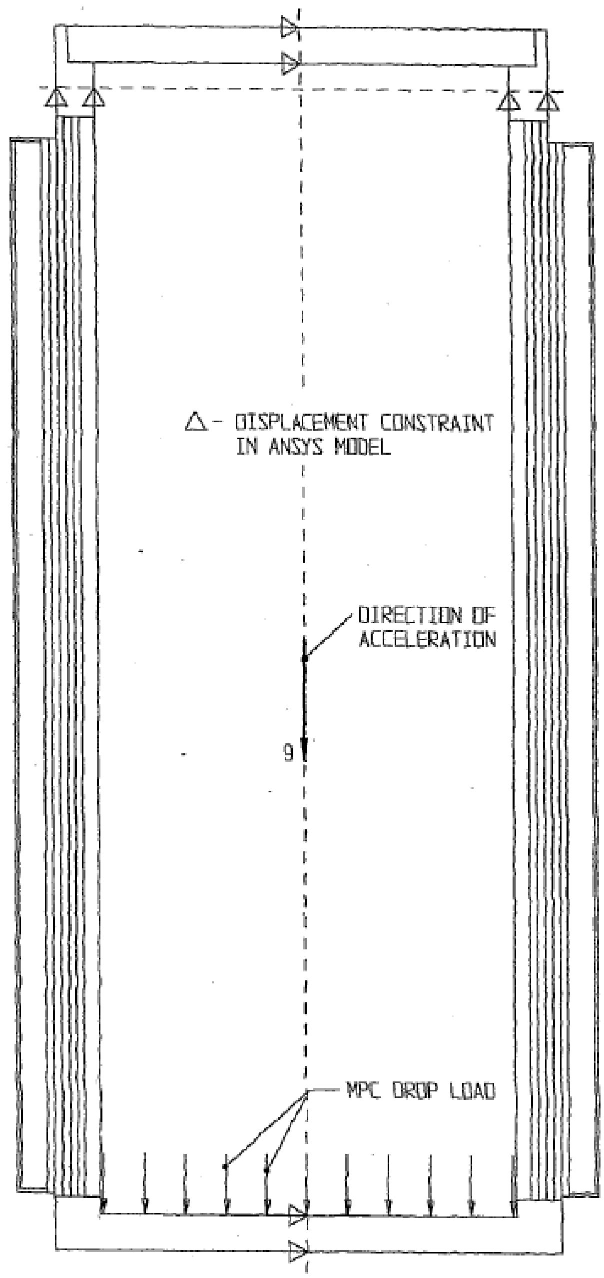

FIGURE 3.4.21; FREE BODY DIAGRAM OF OVERPACK - BOTTOM END DROP

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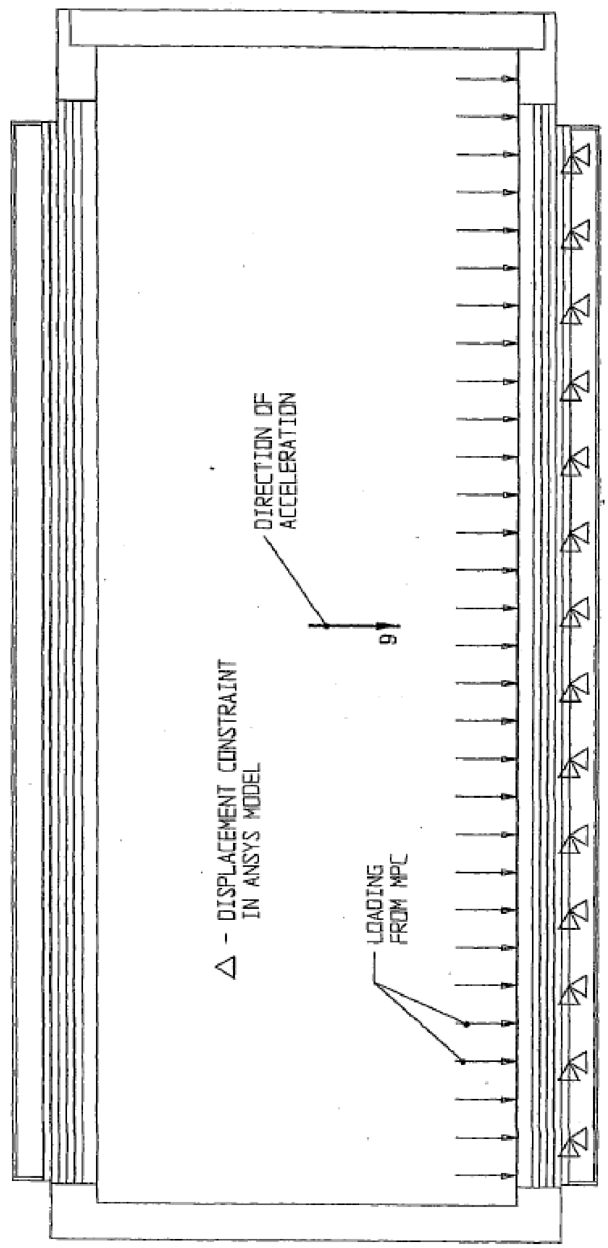

FIGURE 3.4.22; FREE BODY DIAGRAM OF OVERPACK - SIDE DROP

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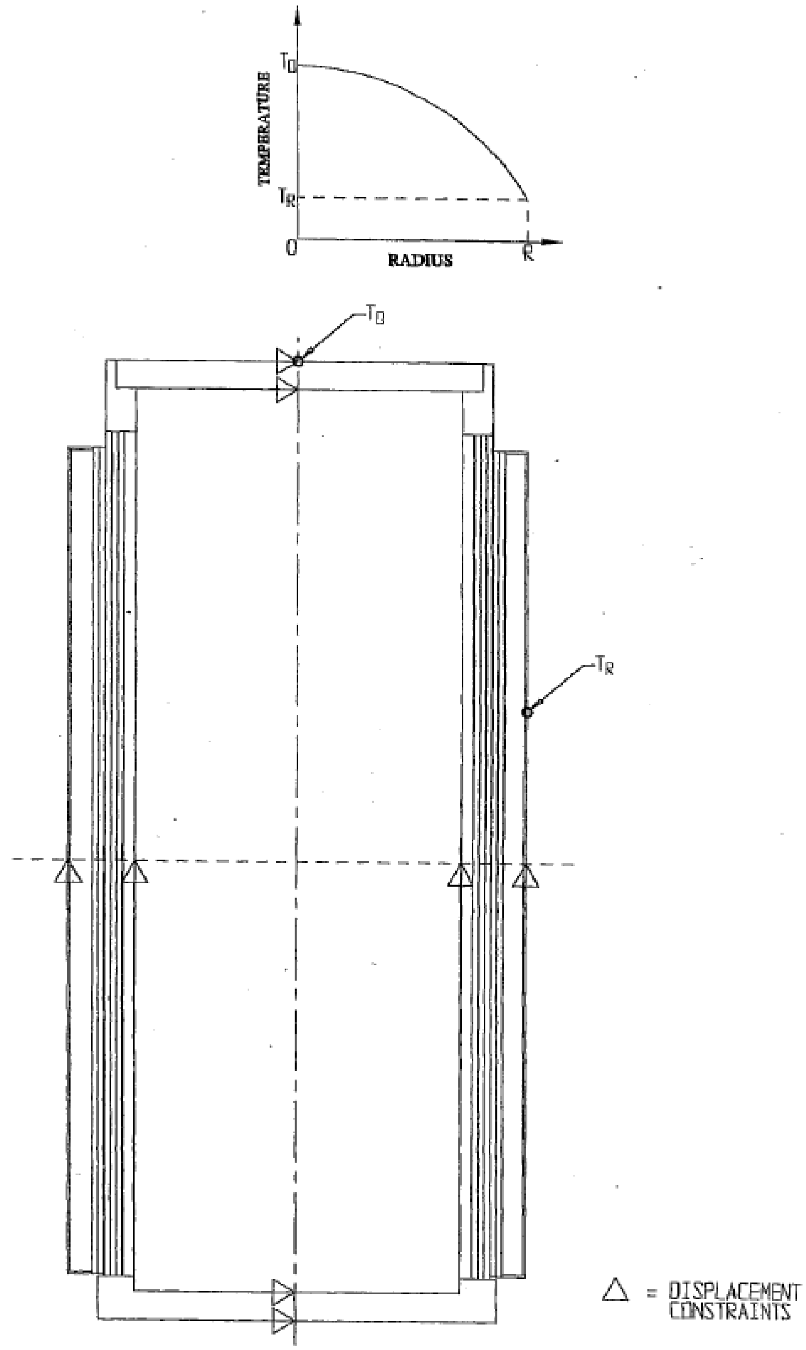

FIGURE 3.4.23; FREE BODY DIAGRAM OF OVERPACK - THERMAL LOAD

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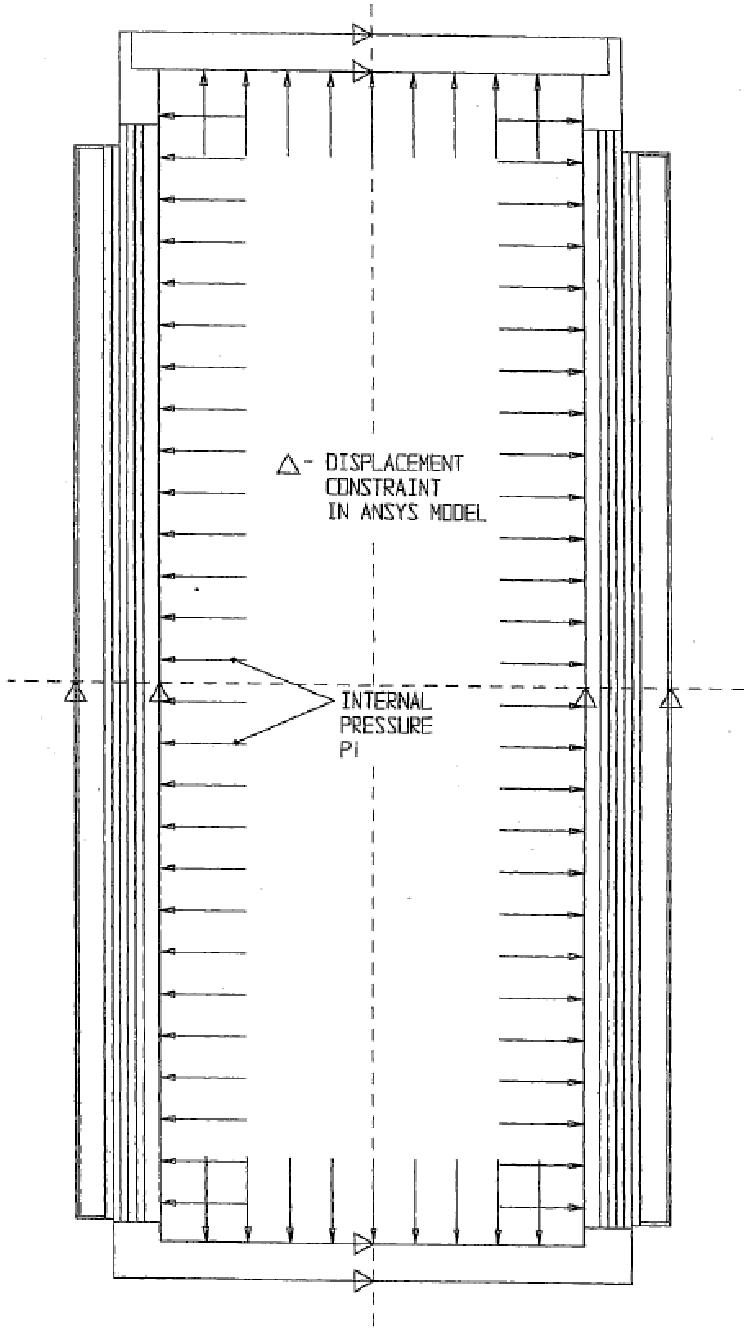

FIGURE 3.4.24; FREE BODY DIAGRAM OF OVERPACK - INTERNAL PRESSURE

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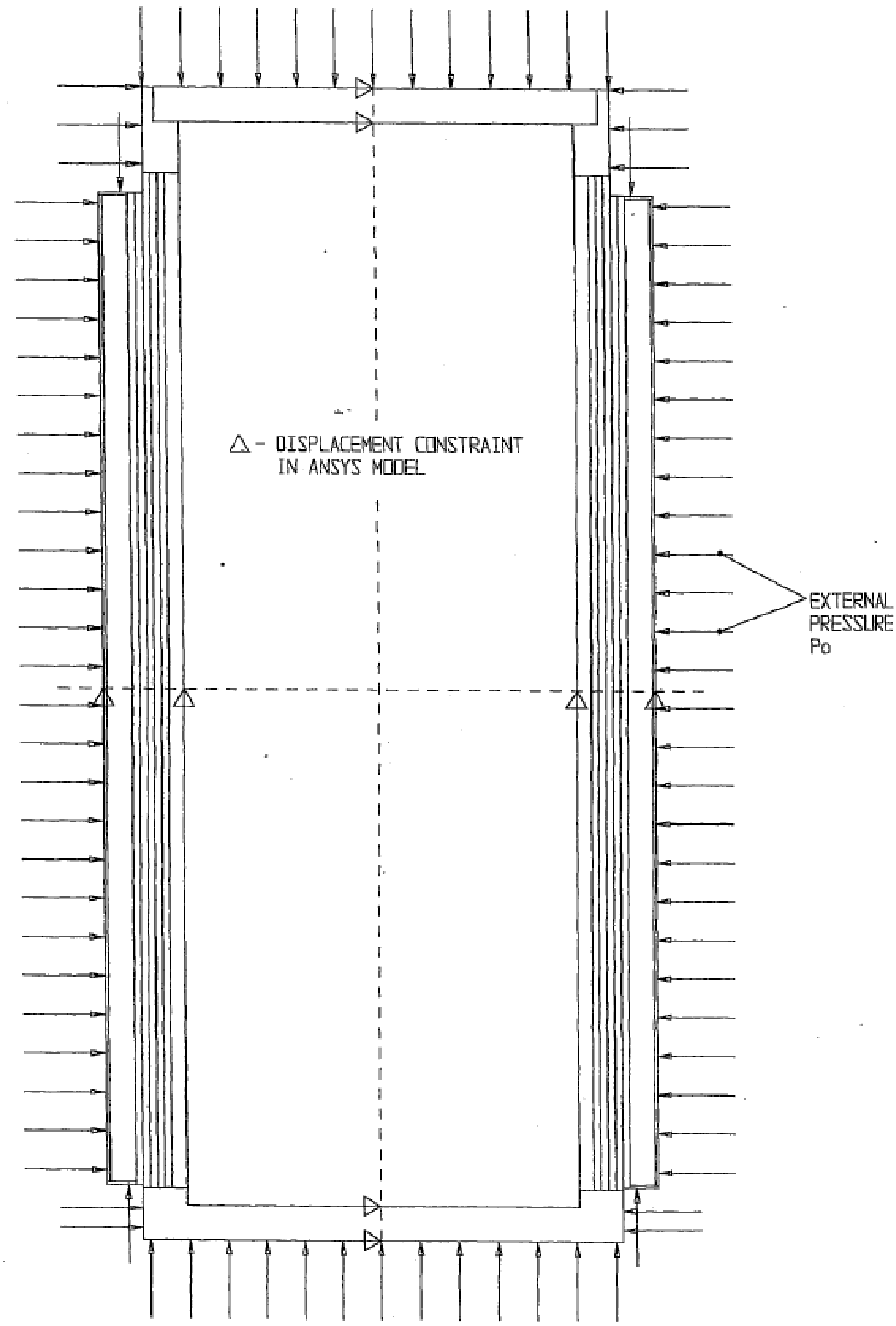

FIGURE 3.4.25; FREE BODY DIAGRAM OF OVERPACK - EXTERNAL PRESSURE

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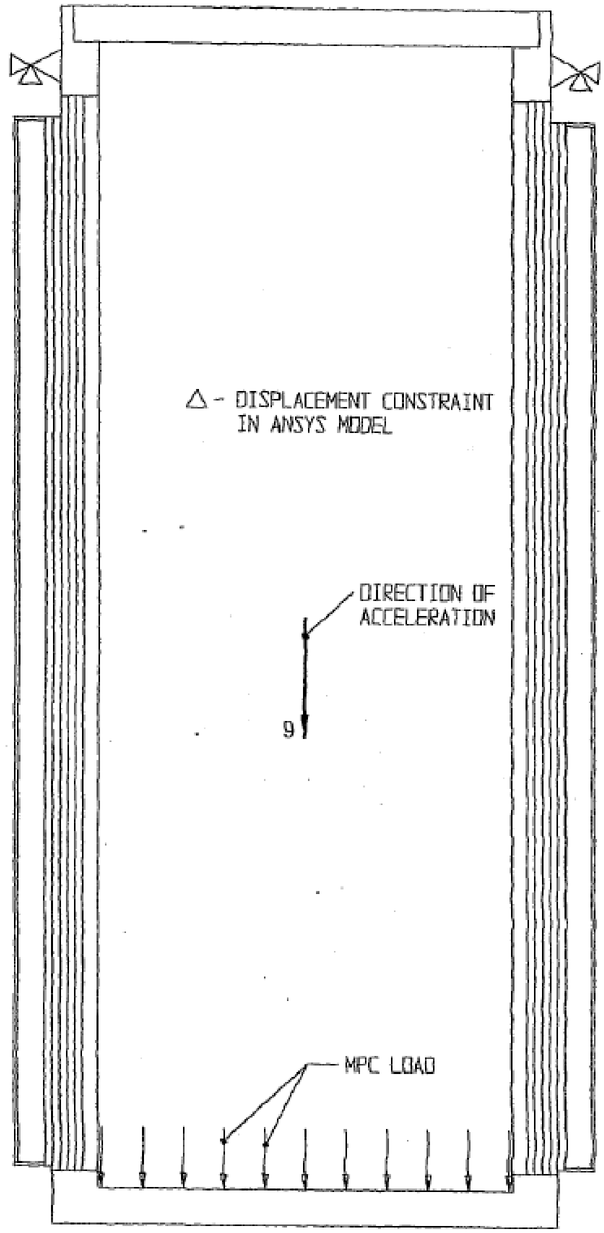

FIGURE 3.4.26; FREE BODY DIAGRAM OF OVERPACK - HANDLING LOAD

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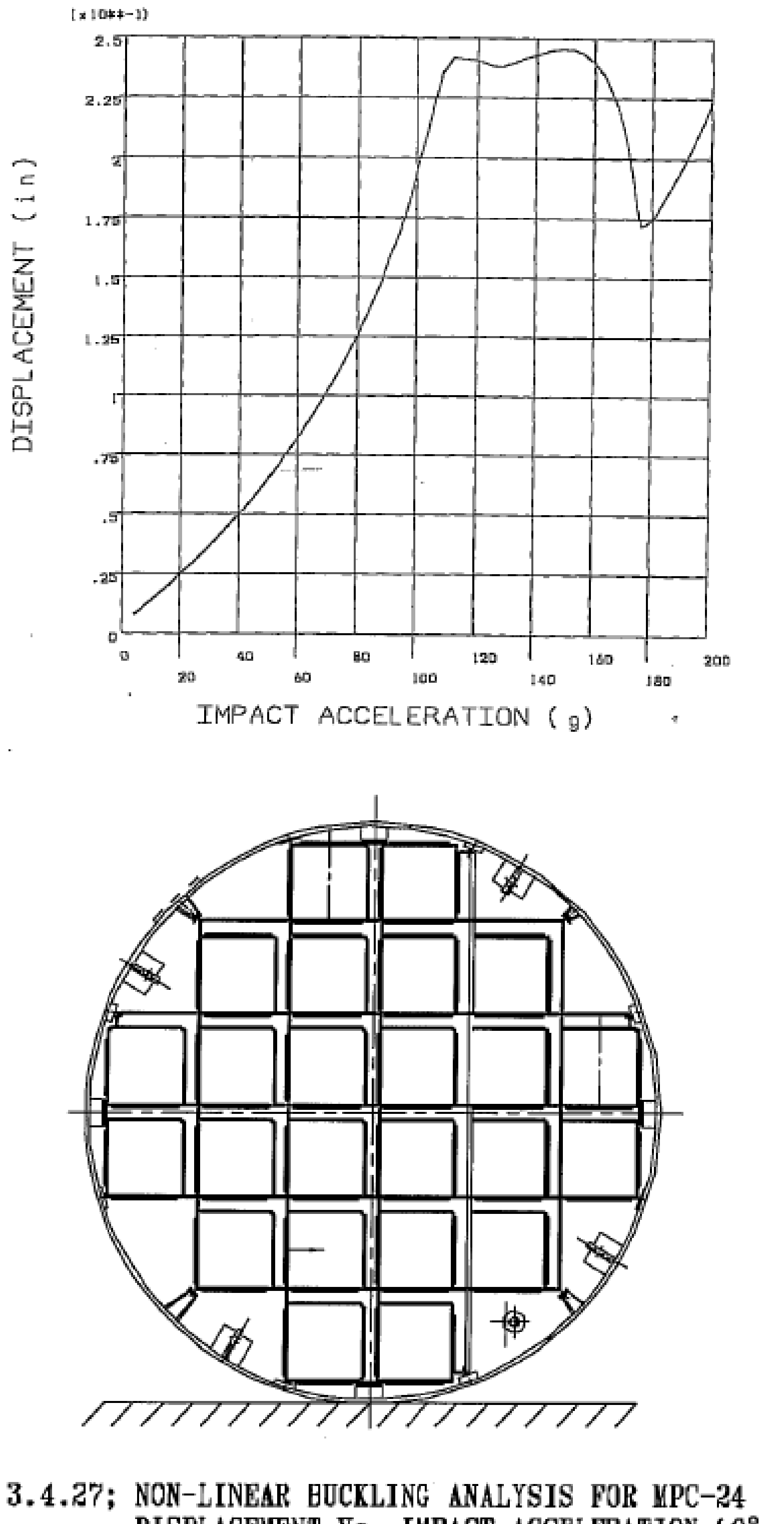

FIGURE 3.4.27; NON-LINEAR BUCKLING ANALYSIS FOR MPC-24
DISPLACEMENT Vs. IMPACT ACCELERATION (0° DROP)

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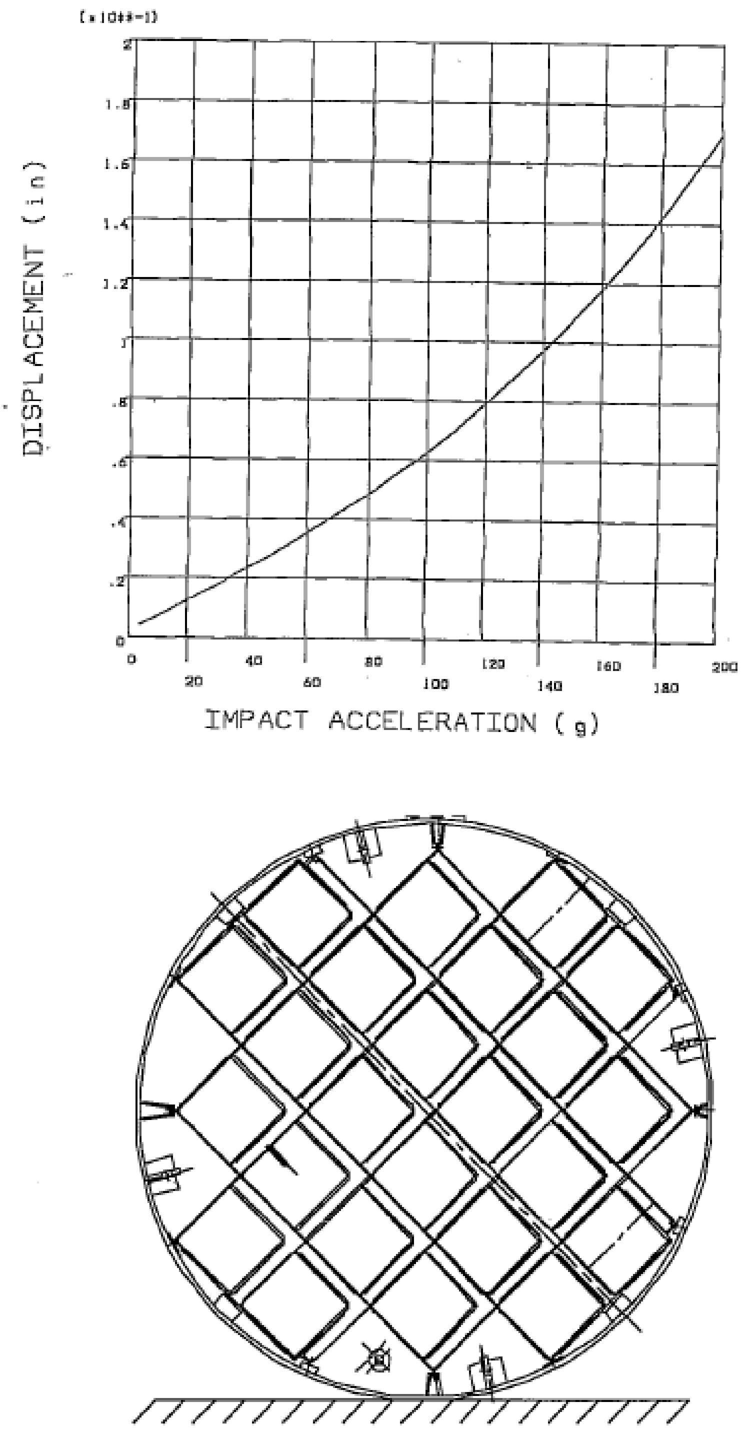

FIGURE 3.4.28; NON-LINEAR BUCKLING ANALYSIS FOR MPC-24
DISPLACEMENT Vs. IMPACT ACCELERATION (45° DROP)

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FIGURE 3.4.29;

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FIGURE 3.4.30;

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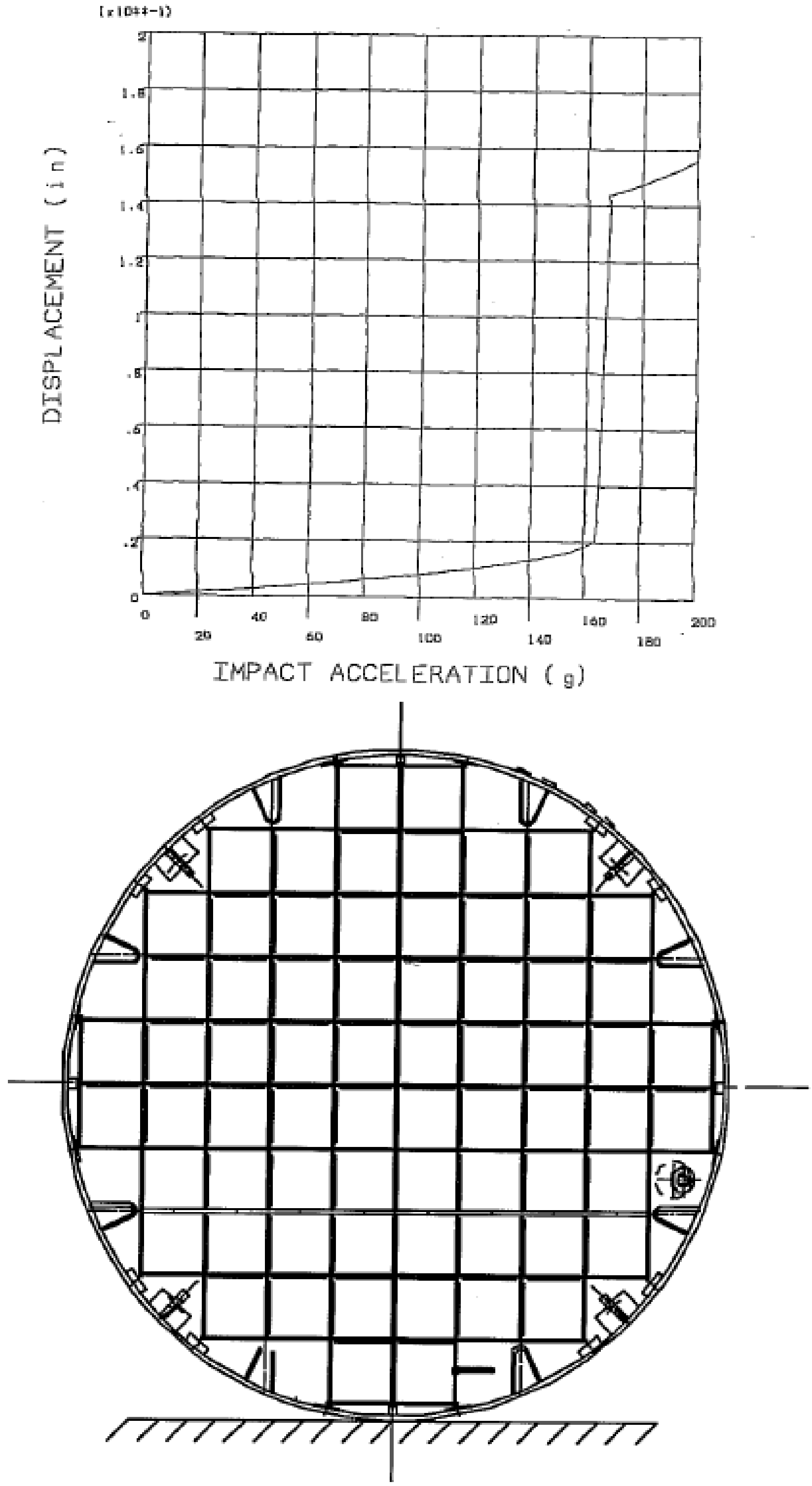

FIGURE 3.4.31; NON-LINEAR BUCKLING ANALYSIS FOR MPC-68 DISPLACEMENT Vs. IMPACT ACCELERATION (0° DROP)

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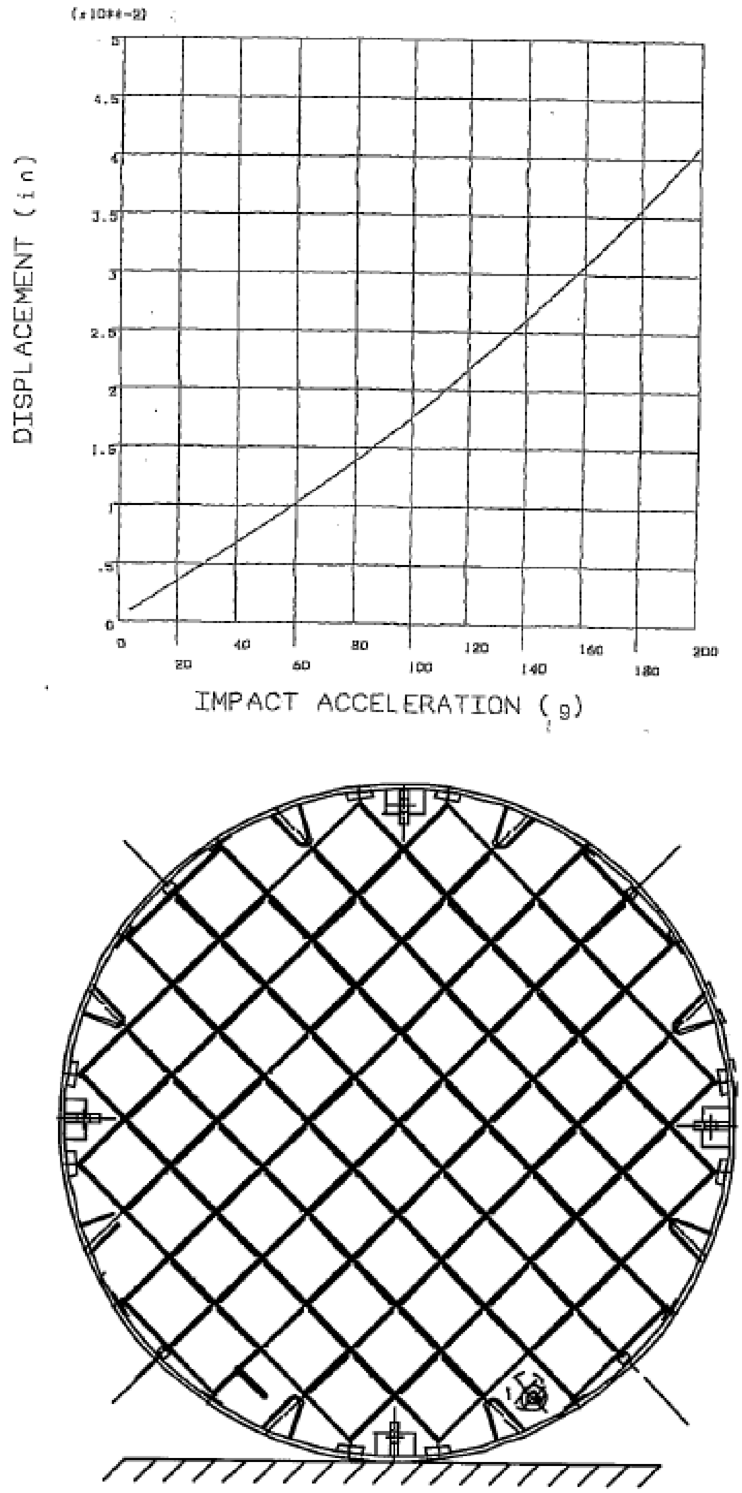

FIGURE 3.4.32; NON-LINEAR BUCKLING ANALYSIS FOR MPC-68 DISPLACEMENT Vs. IMPACT ACCELERATION (45° DROP)

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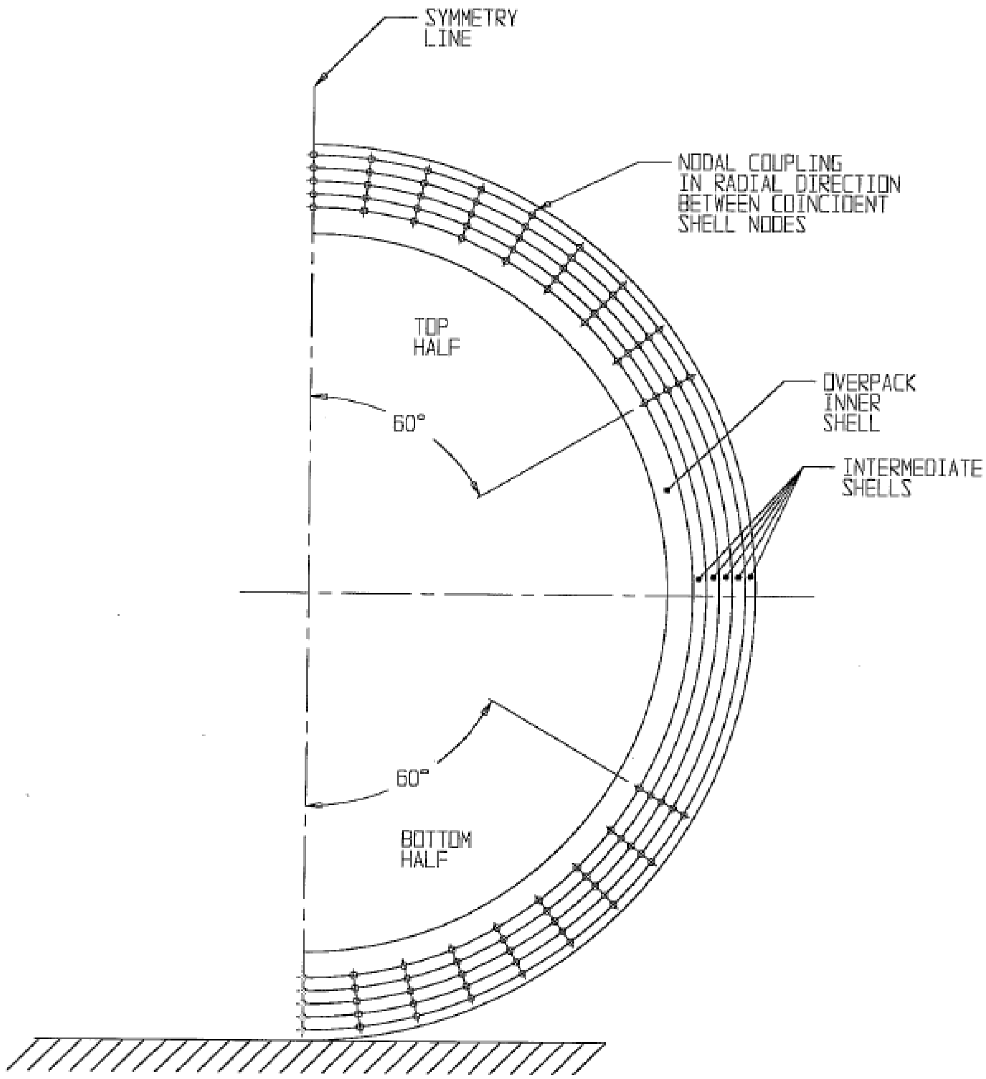

FIGURE 3.4.33; NODAL COUPLING IN OVERPACK
FINITE ELEMENT MODEL

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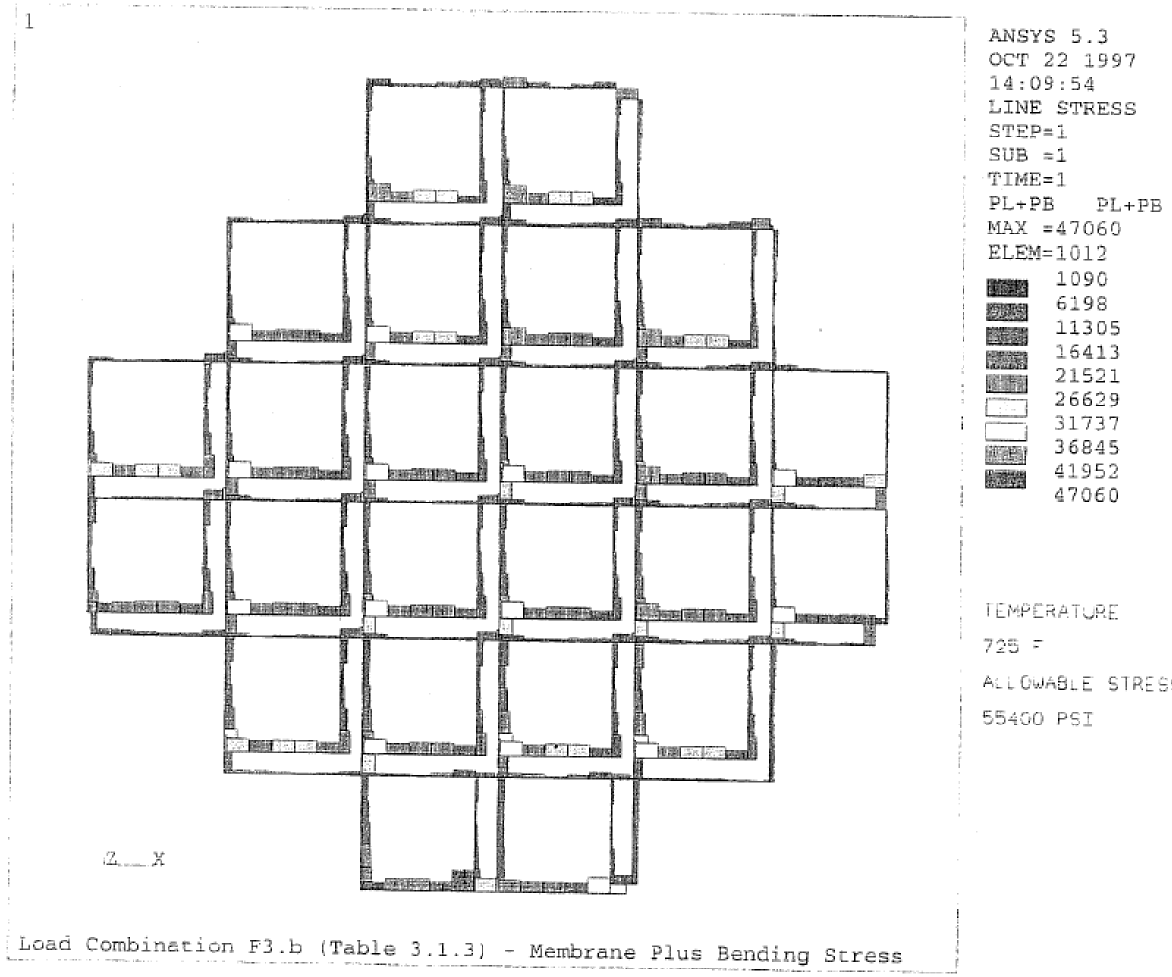

FIGURE 3.4.34; CRITICAL STRESS RESULTS FOR THE MPC-24

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FIGURE 3.4.35;

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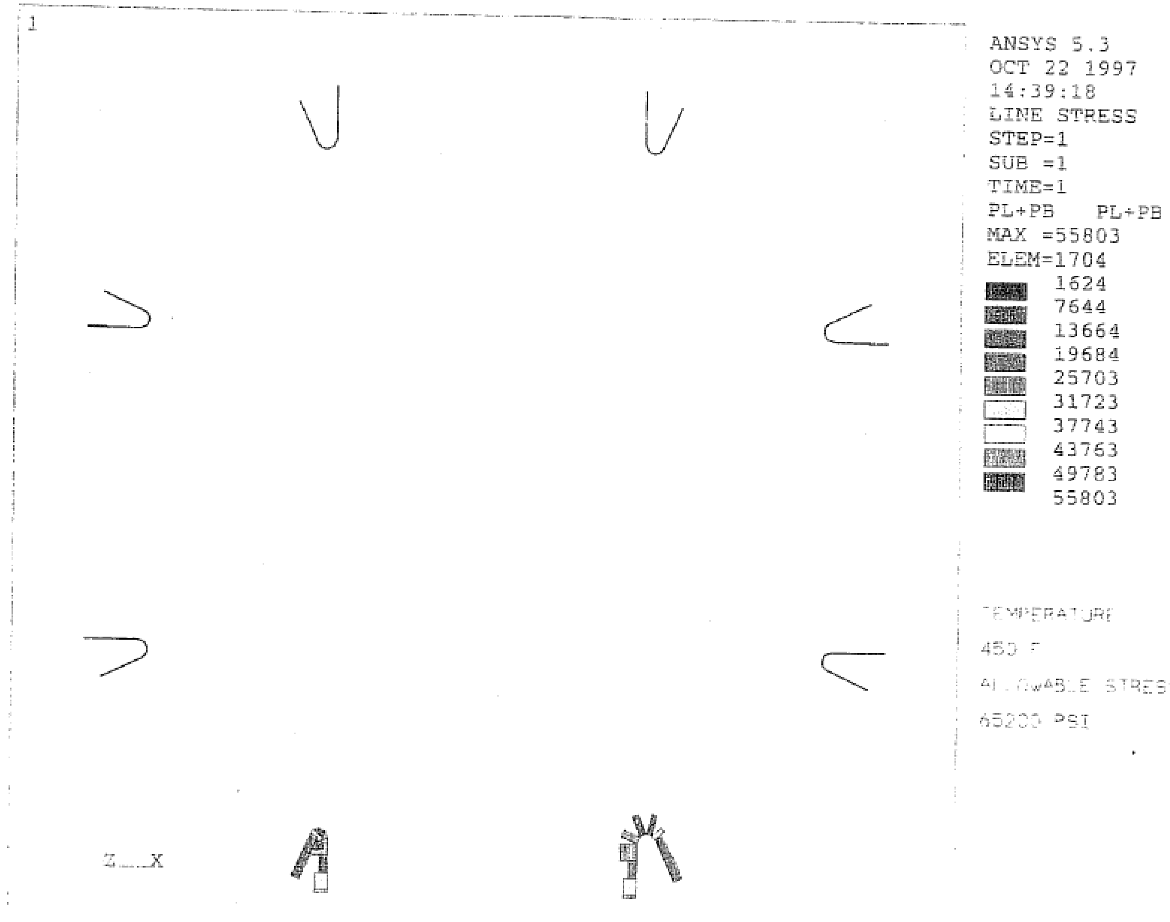

Load Combination E3.b (Table 3.1.3) - Membrane Plus Bending Stress

FIGURE 3.4.36; CRITICAL STRESS RESULTS FOR THE MPC-68

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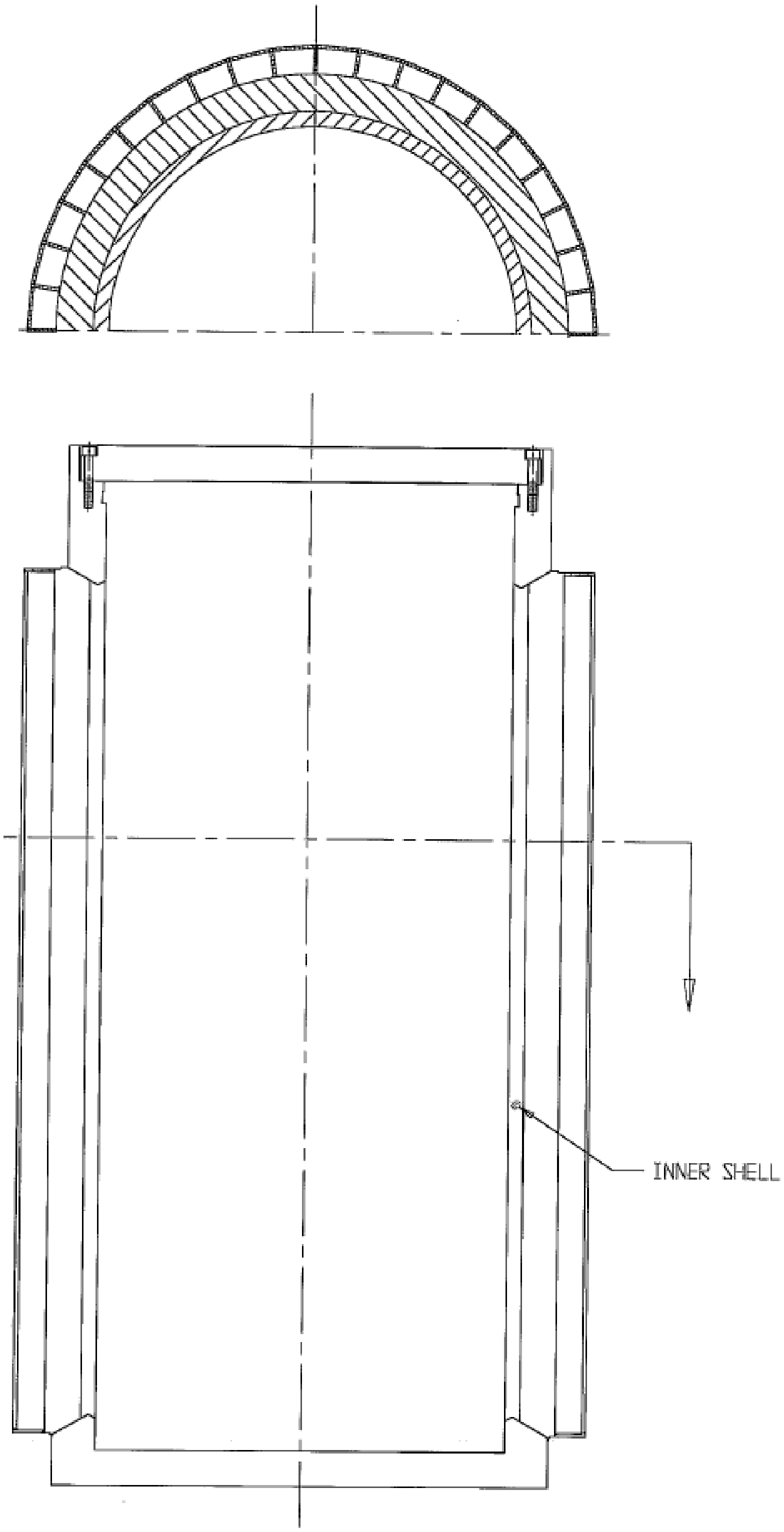

FIGURE 3.4.37; LOCATION OF MINIMUM SAFETY FACTOR FOR LOAD CASE 01

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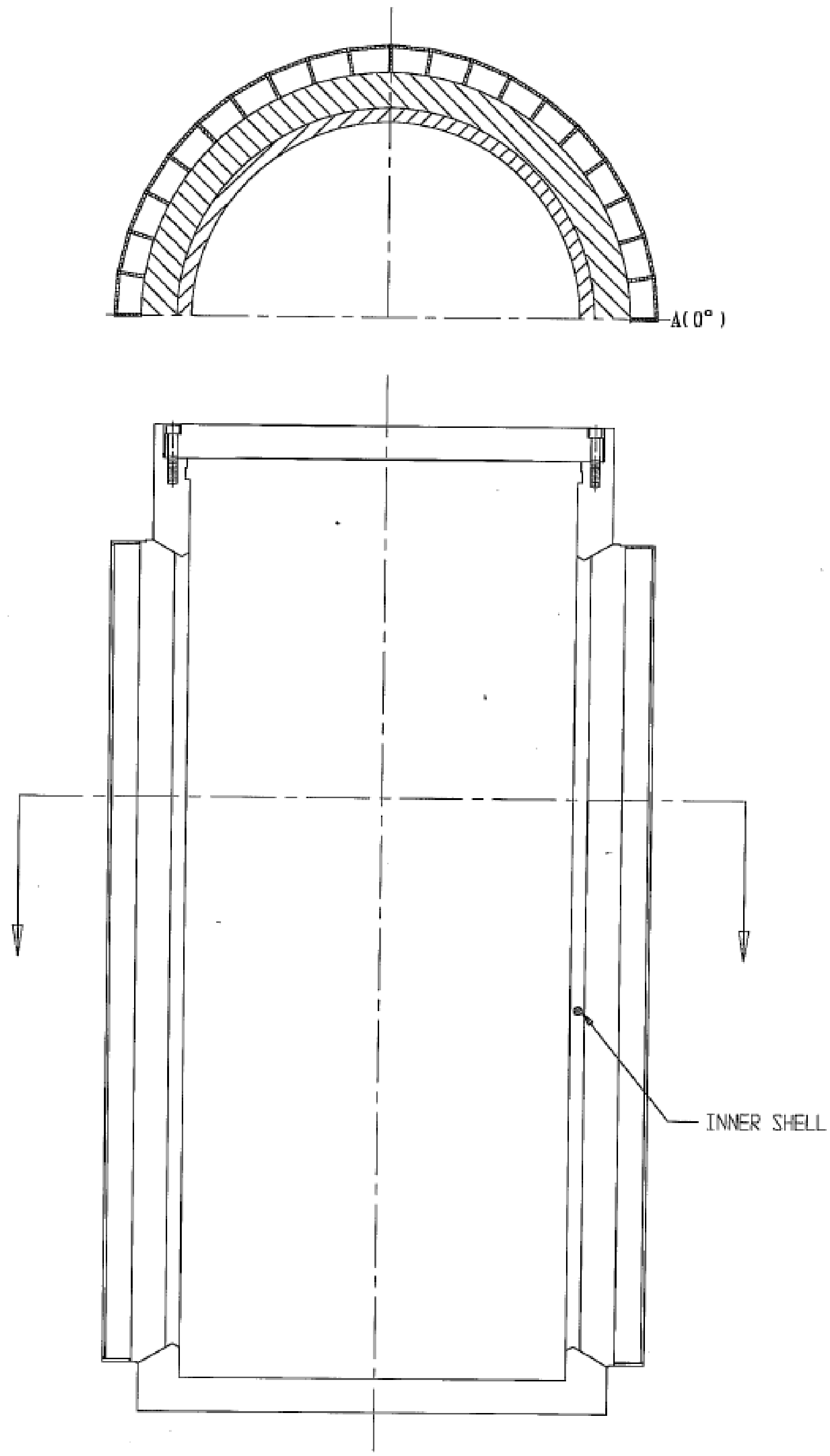

FIGURE 3.4.38; LOCATION OF MINIMUM SAFETY FACTOR FOR LOAD CASE 02

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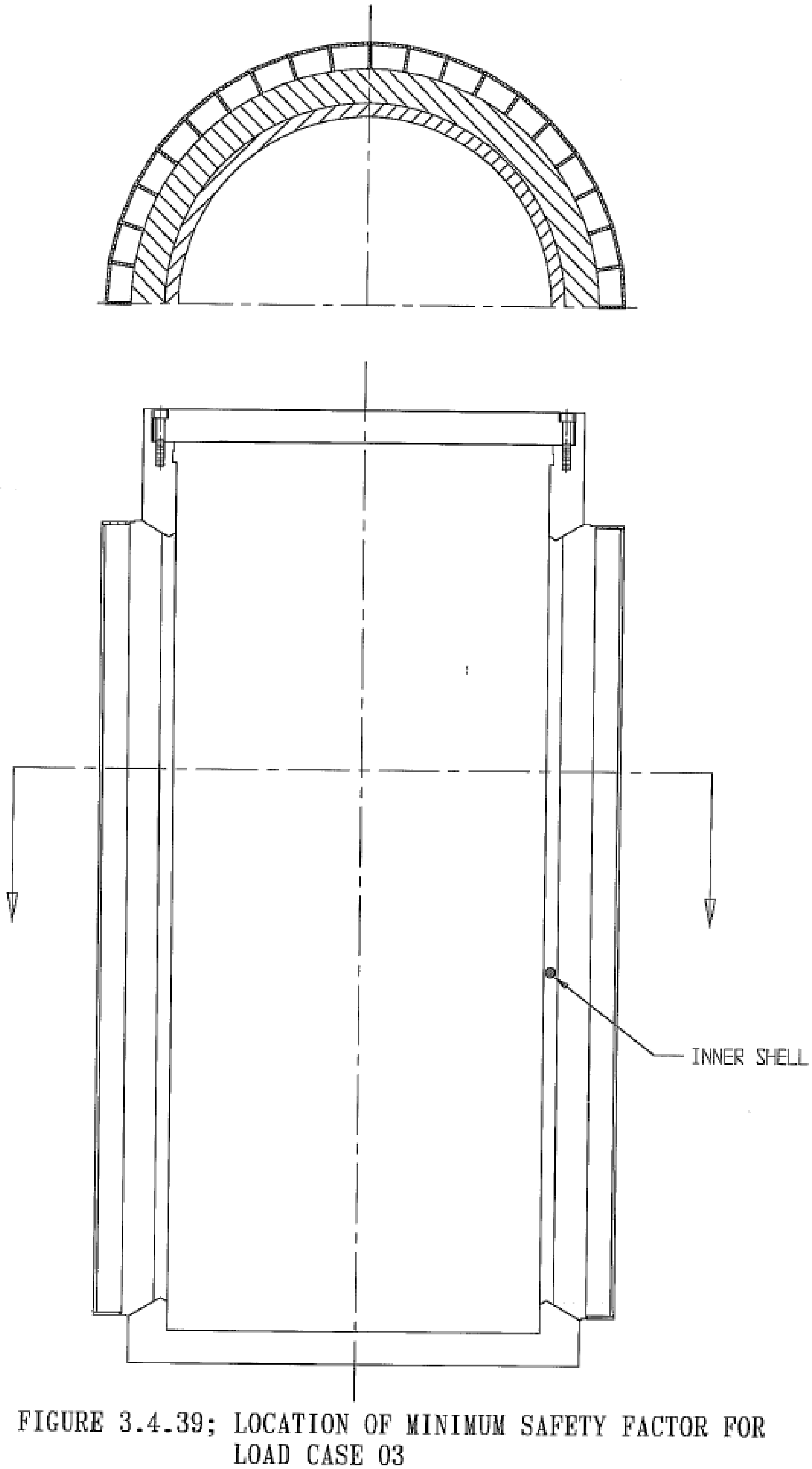

FIGURE 3.4.39; LOCATION OF MINIMUM SAFETY FACTOR FOR
LOAD CASE 03

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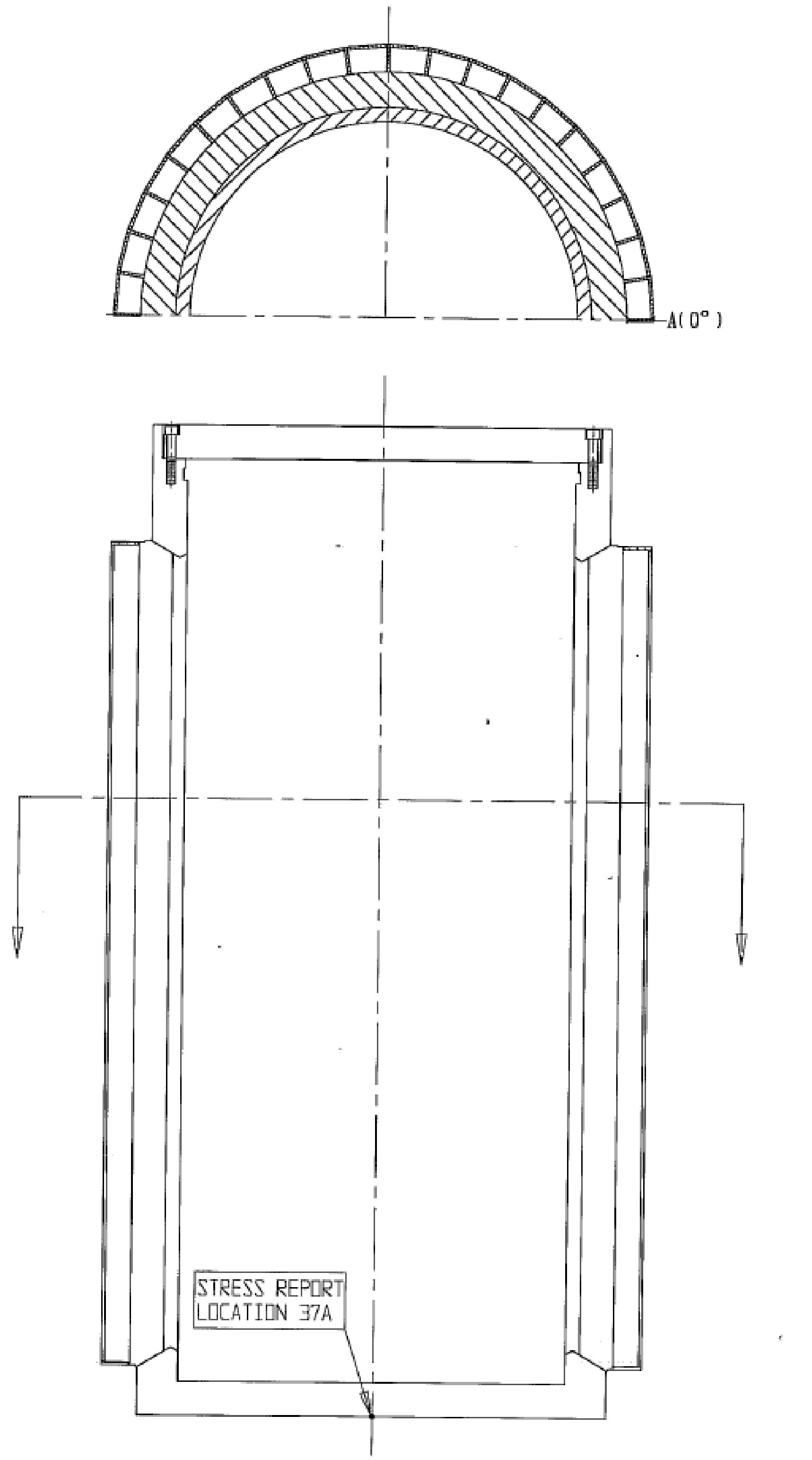

FIGURE 3.4.40; LOCATION OF MINIMUM SAFETY FACTOR FOR LOAD CASE 04.a

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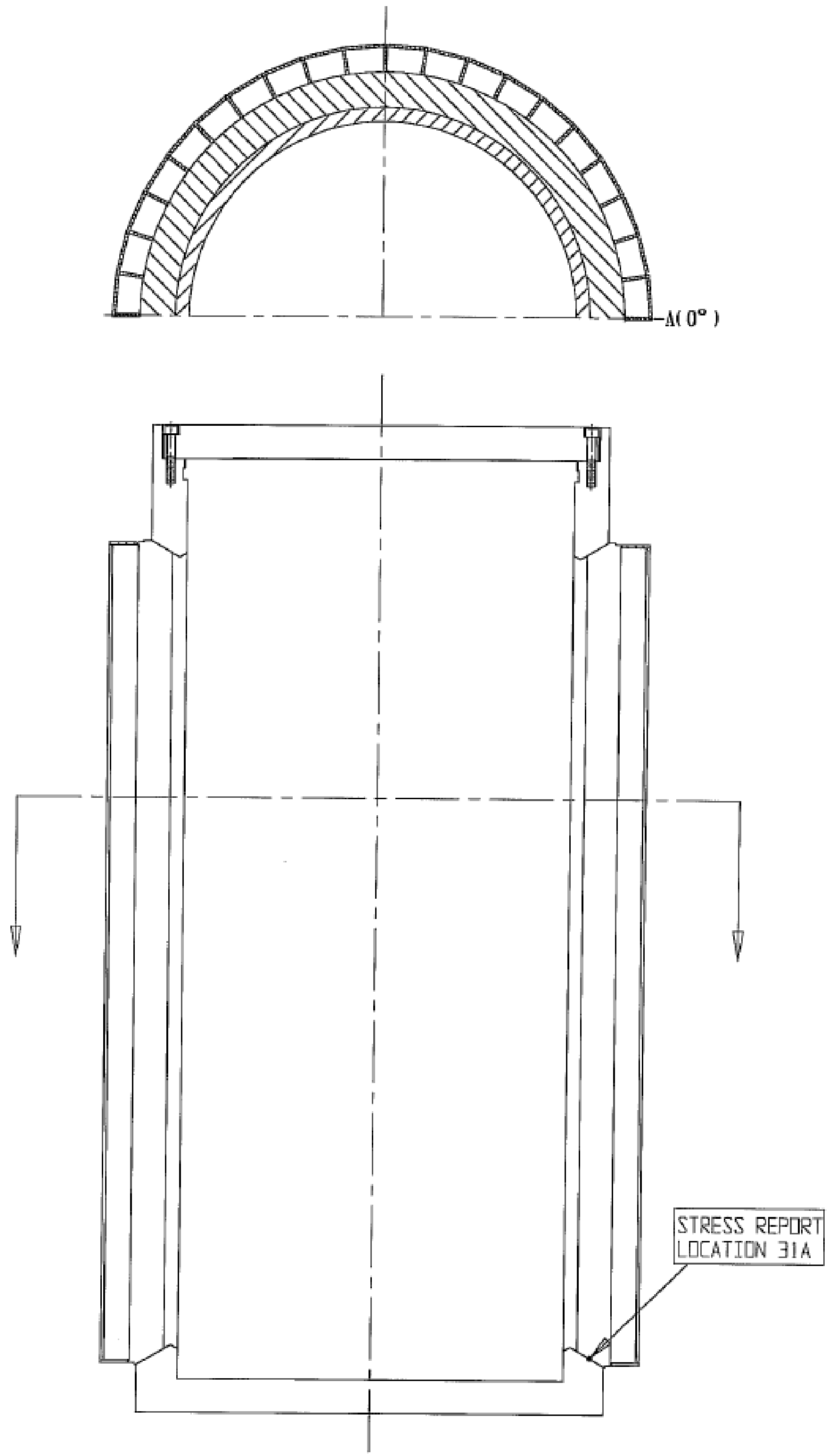

FIGURE 3.4.41; LOCATION OF MINIMUM SAFETY FACTOR FOR
LOAD CASE 04.b

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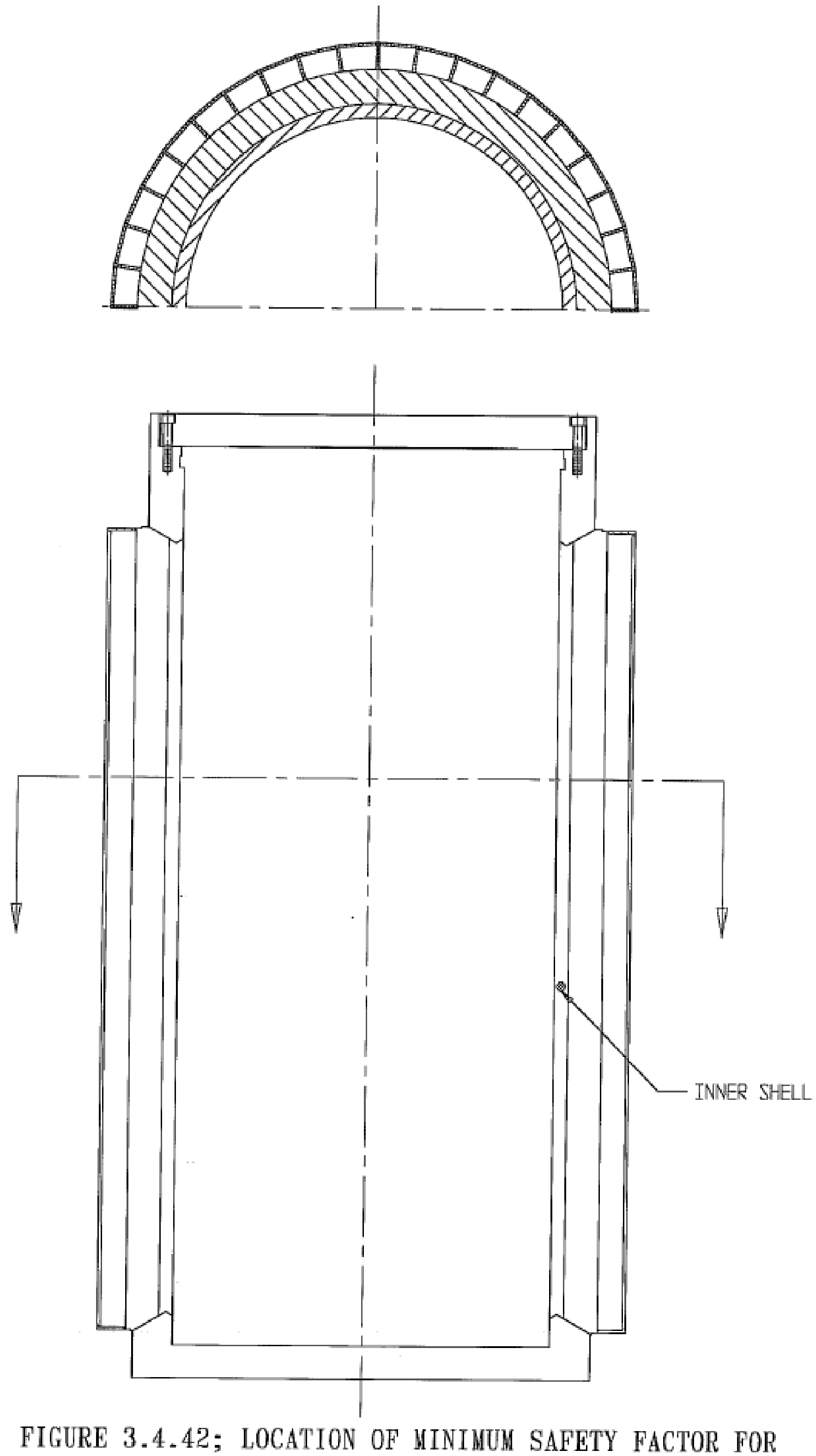

FIGURE 3.4.42; LOCATION OF MINIMUM SAFETY FACTOR FOR LOAD CASE 05

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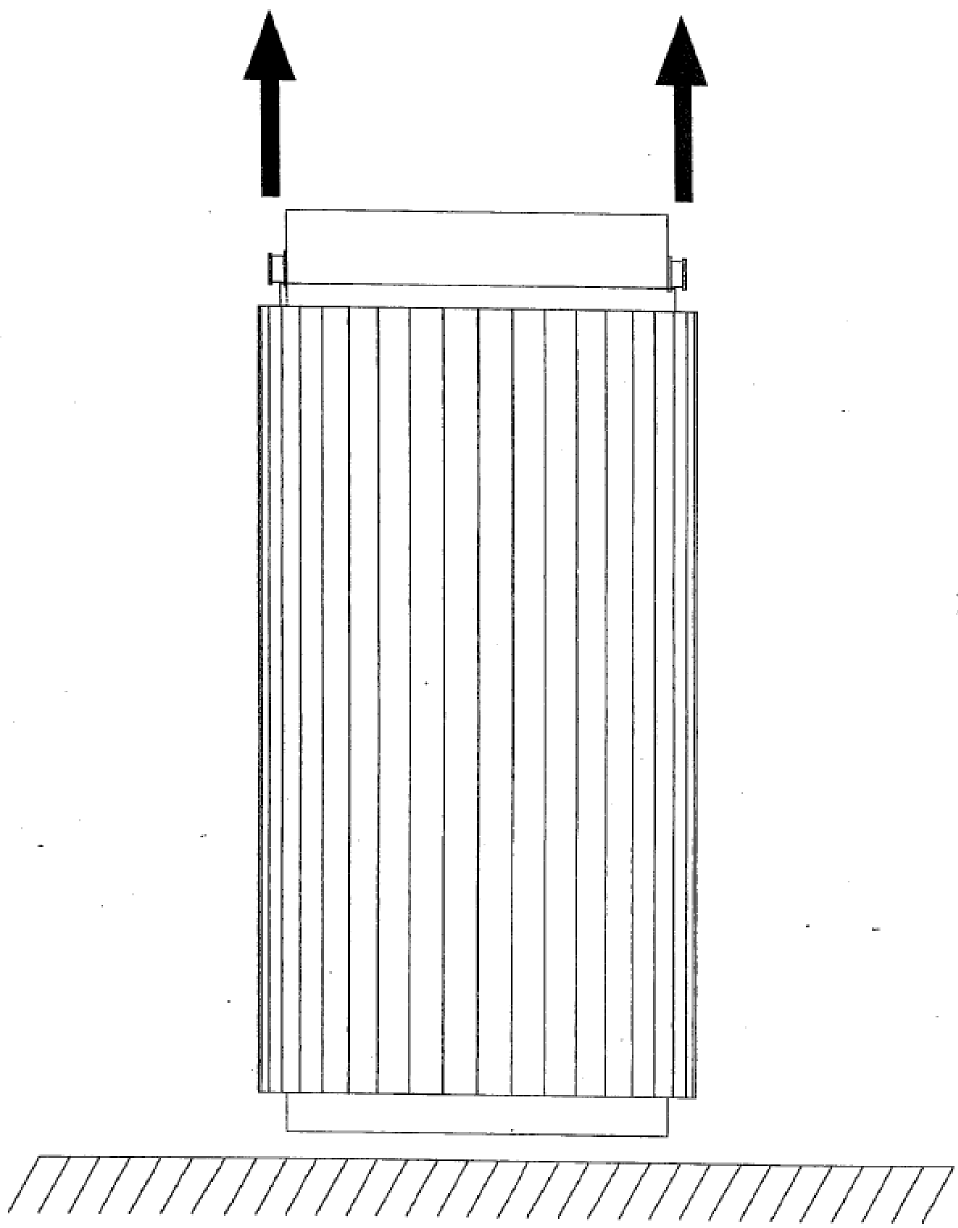

Figure 3.4.43; HI-STAR 100 Vertical Lifting

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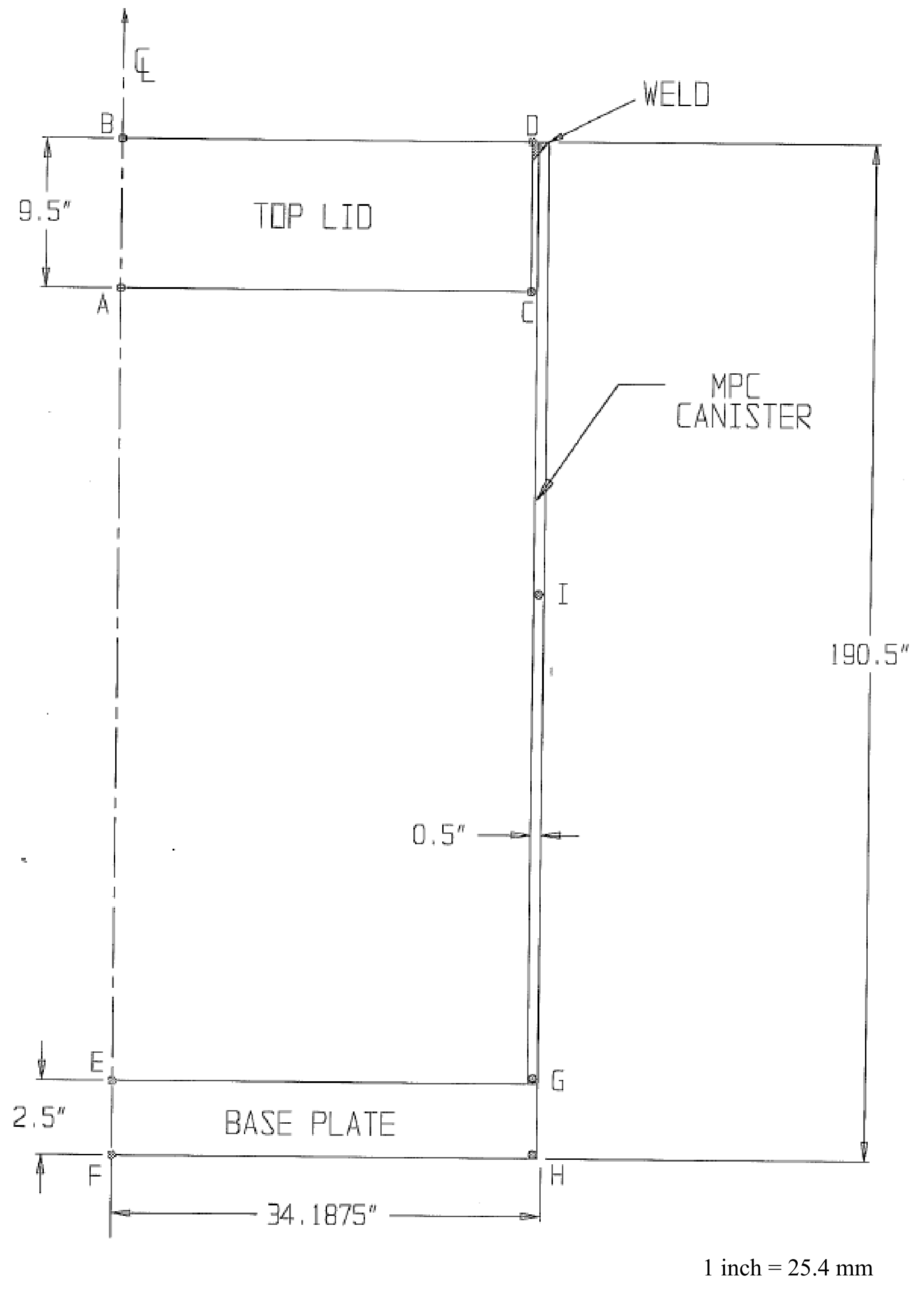

FIGURE 3.4.44 CONFINEMENT BOUNDARY MODEL SHOWING TEMPERATURE DATA POINTS

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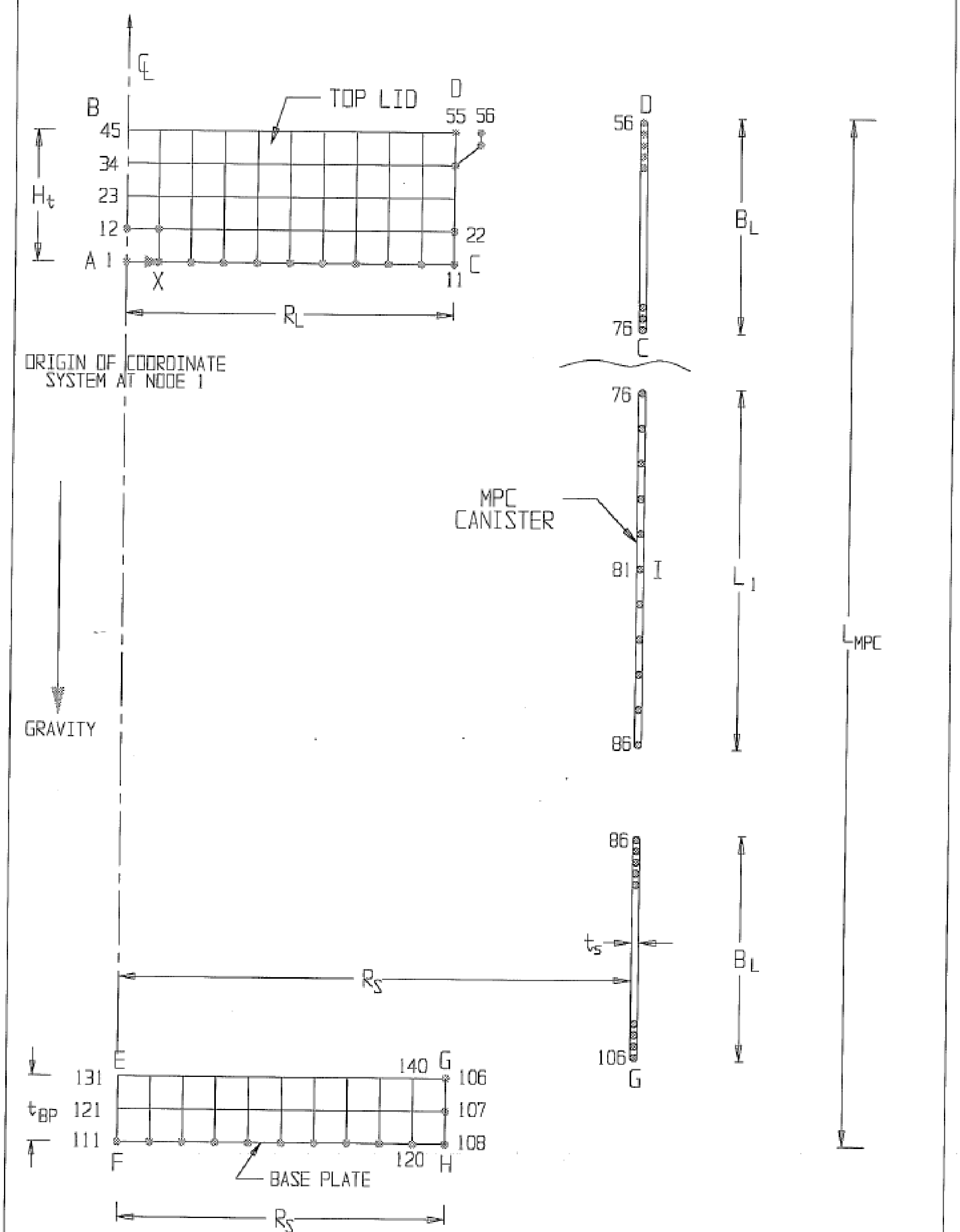

FIGURE 3.4.45 MPC - CONFINEMENT BOUNDARY FINITE ELEMENT GRID (EXPLODED VIEW)

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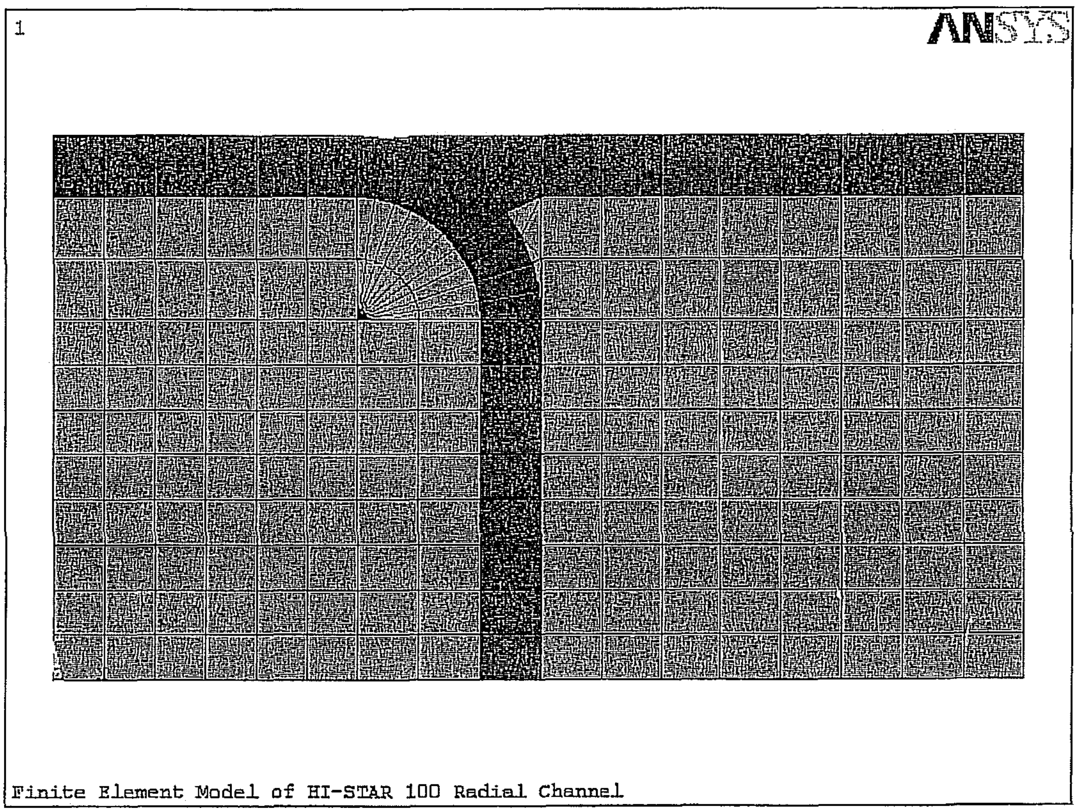

FIGURE 3.4.46; FINITE ELEMENT MODEL OF THERMAL EXPANSION FOAM

3.5 FUEL RODS

The cladding of the fuel rods is the initial confinement boundary in the HI-STAR 100 System. Analyses have been performed in Chapter 4 to ensure that the maximum temperature of the fuel cladding is below the Pacific Northwest Laboratory's threshold values for various cooling times. These temperature limits ensure that the fuel cladding will not degrade in an inert helium environment. Additional details on the fuel rod cladding temperature analyses for the spent fuel to be loaded into the HI-STAR 100 System are provided in Chapter 4.

The dimensions of the storage cell openings in the MPC are equal to or greater than those used in spent fuel racks supplied by Holtec International. Thousands of fuel assemblies have been shuffled in and out of these cells over the years without a single instance of cladding failure. The vast body of physical evidence provides confirmation that the fuel handling and loading operations with the HI-STAR 100 MPC will not endanger or compromise the integrity of the cladding or the structural integrity of the assembly.

The HI-STAR 100 System is designed and evaluated for a maximum deceleration of 60 g's. Studies of the capability of spent fuel rods to resist impact loads [3.5.1] indicate that the most vulnerable fuel can withstand 63 g's in the most adverse orientation. Therefore, designing the HI-STAR 100 System to a maximum deceleration of 60 g's will ensure that fuel rod cladding integrity is maintained during all normal, off-normal, and accident conditions.

3.6 SUPPLEMENTAL DATA

3.6.1 Additional Codes and Standards Referenced in HI-STAR 100 System Design and Fabrication

The following additional codes, standards and practices were used as aids in developing the design, manufacturing, quality control and testing methods for HI-STAR 100 System:

a. Design Codes

(1) AISC Manual of Steel Construction, 1964 Edition and later.

(2) ANSI N210-1976, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations".

(3) American Concrete Institute Building Code Requirements for Reinforced Concrete, ACI-318.

(4) Code Requirements for Nuclear Safety Related Concrete Structures, ACI349-85/ACI349R-85, and ACI349.1R-80.

(5) ASME NQA-1, Quality Assurance Program Requirements for Nuclear Facilities.

(6) ASME NQA-2-1989, Quality Assurance Requirements for Nuclear Facility Applications.

(7) ANSI Y14.5M, Dimensioning and Tolerancing for Engineering Drawings and Related Documentation Practices.

(8) ACI Detailing Manual - 1980.

(9) Crane Manufacturer's Association of America, Inc., CMAA Specification #70, Specifications for Electric Overhead Traveling Cranes, Revised 1988.

b. Material Codes - Standards of ASTM

(1) E165 - Standard Methods for Liquid Penetrant Inspection.

(2) A240 - Standard Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet and Strip for Fusion-Welded Unfired Pressure Vessels.

(3) A262 - Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steel.

(4) A276 - Standard Specification for Stainless and Heat-Resisting Steel Bars and Shapes.

(5) A479 - Steel Bars for Boilers & Pressure Vessels.

(6) ASTM A564, Standard Specification for Hot-Rolled and Cold-Finished Age-Hardening Stainless and Heat-Resisting Steel Bars and Shapes.

(7) C750 - Standard Specification for Nuclear-Grade Boron Carbide Powder.

(8) A380 - Recommended Practice for Descaling, Cleaning and Marking Stainless Steel Parts and Equipment.

(9) C992 - Standard Specification for Boron-Based Neutron Absorbing Material Systems for Use in Nuclear Spent Fuel Storage Racks.

(10) ASTM E3, Preparation of Metallographic Specimens.

(11) ASTM E190, Guided Bend Test for Ductility of Welds.

(12) NCA3800 - Metallic Material Manufacturer's and Material Supplier's Quality System Program.

c. Welding Codes: ASME Boiler and Pressure Vessel Code, Section IX - Welding and Brazing Qualifications, 1995 Edition.

d. Quality Assurance, Cleanliness, Packaging, Shipping, Receiving, Storage, and Handling Requirements

(1) ANSI 45.2.1 - Cleaning of Fluid Systems and Associated Components during Construction Phase of Nuclear Power Plants.

(2) ANSI N45.2.2 - Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants (During the Construction Phase).

(3) ANSI - N45.2.6 - Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants (Regulatory Guide 1.58).

(4) ANSI-N45.2.8, Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants.

(5) ANSI - N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants.

(6) ANSI-N45.2.12, Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants.

(7) ANSI N45.2.13 - Quality Assurance Requirements for Control of Procurement of Equipment Materials and Services for Nuclear Power Plants (Regulatory Guide 1.123).

(8) ANSI N45.2.15-18 - Hoisting, Rigging, and Transporting of Items for Nuclear Power Plants.

(9) ANSI N45.2.23 - Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants (Regulatory Guide 1.146).

(10) ASME Boiler and Pressure Vessel, Section V, Nondestructive Examination, 1995 Edition.

(11) ANSI - N16.9-75 Validation of Calculation Methods for Nuclear Criticality Safety.

e. Reference NRC Design Documents

(1) NUREG-0800, Radiological Consequences of Fuel Handling Accidents.

(2) NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants", USNRC, Washington, D.C., July, 1980.

(3) NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", USNRC, January 1997, Final Report.

f. Other ANSI Standards (not listed in the preceding)

(1) ANSI/ANS 8.1 (N16.1) - Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.

(2) ANSI/ANS 8.17, Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors.

(3) N45.2 - Quality Assurance Program Requirements for Nuclear Facilities - 1971.

(4) N45.2.9 - Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants - 1974.

(5) N45.2.10 - Quality Assurance Terms and Definitions - 1973.

(6) ANSI/ANS 57.2 (N210) - Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants.

(7) N14.6 (1993) - American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or more for Nuclear Materials.

(8) ANSI/ASME N626-3, Qualification and Duties of Personnel Engaged in ASME Boiler and Pressure Vessel Code Section III, Div. 1, Certifying Activities.

g. Code of Federal Regulations

(1) 10CFR20 - Standards for Protection Against Radiation.

(2) 10CFR21 - Reporting of Defects and Non-compliance.

(3) 10CFR50 - Appendix A - General Design Criteria for Nuclear Power Plants.

(4) 10CFR50 - Appendix B - Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

(5) 10CFR61 - Licensing Requirements for Land Disposal of Radioactive Material.

(6) 10CFR71 - Packaging and Transportation of Radioactive Material.

h. Regulatory Guides

(1) RG 1.13 - Spent Fuel Storage Facility Design Basis (Revision 2 Proposed).

(2) RG 1.25 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility of Boiling and Pressurized Water Reactors.

(3) RG 1.28 - (ANSI N45.2) - Quality Assurance Program Requirements.

(4) RG 1.29 - Seismic Design Classification (Rev. 3).

(5) RG 1.31 - Control of Ferrite Content in Stainless Steel Weld Material.

(6) RG 1.38 - (ANSI N45.2.2) Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants.

(7) RG 1.44 - Control of the Use of Sensitized Stainless Steel.

(8) RG 1.58 - (ANSI N45.2.6) Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel.

(9) RG 1.61 - Damping Values for Seismic Design of Nuclear Power Plants, Rev. 0, 1973.

(10) RG 1.64 - (ANSI N45.2.11) Quality Assurance Requirements for the Design of Nuclear Power Plants.

(11) RG 1.71 - Welder Qualifications for Areas of Limited Accessibility.

(12) RG 1.74 - (ANSI N45.2.10) Quality Assurance Terms and Definitions.

(13) RG 1.85 - Materials Code Case Acceptability - ASME Section 3, Div. 1.

(14) RG 1.88 - (ANSI N45.2.9) Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records.

(15) RG 1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis.

(16) RG 1.122 - Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components.

(17) RG 1.123 - (ANSI N45.2.13) Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants.

(18) RG 1.124 - Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports, Revision 1, 1978.

(19) RG 3.4 - Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities.

(20) RG 3.41 - Validation of Calculational Methods for Nuclear Criticality Safety, Revision 1, 1977.

(21) RG 8.8 - Information Relative to Ensuring that Occupational Radiation Exposure at Nuclear Power Plants will be as Low as Reasonably Achievable (ALARA).

(22) DG-8006, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants".

i. Branch Technical Position

(1) CPB 9.1-1 - Criticality in Fuel Storage Facilities.

(2) ASB 9-2 - Residual Decay Energy for Light-Water Reactors for Long-Term Cooling.

j. Standard Review Plan (NUREG-0800)

(1) SRP 3.2.1 - Seismic Classification.

(2) SRP 3.2.2 - System Quality Group Classification.

(3) SRP 3.7.1 - Seismic Design Parameters.

(4) SRP 3.7.2 - Seismic System Analysis.

(5) SRP 3.7.3 - Seismic Subsystem Analysis.

(6) SRP 3.8.4 - Other Seismic Category I Structures (including Appendix D), Technical Position on Spent Fuel Rack.

(7) SRP 3.8.5 - Foundations

(8) SRP 9.1.2 - Spent Fuel Storage, Revision 3, 1981.

(9) SRP 9.1.3 - Spent Fuel Pool Cooling and Cleanup System.

(10) SRP 9.1.4 - Light Load Handling System.

(11) SRP 9.1.5 - Overhead Heavy Load Handling System.

(12) SRP 15.7.4 - Radiological Consequences of Fuel Handling Accidents.

k. AWS Standards

(1) AWS D1.1 - Structural Welding Code, Steel.

(2) AWS A2.4 - Standard Symbols for Welding, Brazing and Nondestructive Examination.

(3) AWS A3.0 - Standard Welding Terms and Definitions.

(4) AWS A5.12 - Tungsten Arc-welding Electrodes.

(5) AWS QC1 - Standards and Guide for Qualification and Certification of Welding Inspectors.

l. Others

(1) ASNT-TC-1A - Recommended Practice for Nondestructive Personnel Qualification and Certification.

(2) SSPC SP-2 - Surface Preparation Specification No. 2 Hand Tool Cleaning.

(3) SSPC SP-3 - Surface Preparation Specification No. 3 Power Tool Cleaning.

(4) SSPC SP-10 - Near-White Blast Cleaning.

3.6.2 Computer Programs

Three computer programs, all with a well established history of usage in the nuclear industry, have been utilized to perform structural and mechanical analyses documented in this report. These codes are ANSYS, DYNA3D, and WORKING MODEL. ANSYS is a public domain code which utilizes the finite element method for structural analyses.

WORKING MODEL, Version V.3.0

This code is primarily used in the analysis of cask drop events for the 10CFR71 submittal. It is also used in this 10CFR72 submittal to verify the results reported in Appendix 3.A for tip-over and to compute the dynamic load resulting from intermediate missile impact on the overpack closure in Supplement 22 of [3.4.13].

"WORKING MODEL" (V3.0) is a Computer Aided Engineering (CAE) tool with an integrated user interface that merges modeling, simulation, viewing, and measuring. The program includes a dynamics algorithm that provides automatic collision and contact handling, including detection, response, restitution, and friction.

Numerical integration is performed using the Kutta-Merson integrator which offers options for variable or fixed time-step and error bounding.

The Working Model Code is commercially available. Holtec has performed independent QA validation of the code by comparing the solution of several classical dynamics problems with the numerical results predicted by Working Model. Agreement in all cases is excellent.

Additional theoretical material is available in the manual: "Users Manual, Working Model, Version 3", Knowledge Revolution, 66 Bovet Road, Suite 200, San Mateo, CA, 94402.

DYNA3D

"DYNA3D" is a nonlinear, explicit, three-dimensional finite element code for solid and structural mechanics. It was originally developed at Lawrence Livermore Laboratories and is ideally suited for study of short-time duration, highly nonlinear impact problems in solid mechanics. DYNA3D is commercially available for both UNIX work stations and Pentium class PCs running Windows 95 or Windows NT. The PC version has been fully validated at Holtec following Holtec's QA procedures for commercial computer codes. This code is used to analyze the drop accidents and the tip-over scenario for the HI-STAR 100. Benchmarking of DYNA3D for these storage analyses is discussed and documented in Appendix 3.A.

3.6.3 Appendices Included in Chapter 3

3.A HI-STAR Deceleration Under Postulated Drop Events and Tipover

3.7 COMPLIANCE TO NUREG-1536

Supporting information to provide reasonable assurance with respect to the adequacy of the HI-STAR 100 System to store spent nuclear fuel in accordance with the stipulations of the Technical Specifications (Chapter 12) is provided throughout this Final Safety Analysis Report. An itemized table (Table 3.0.1 at the beginning of this chapter) has been provided to locate and collate the substantiating material to support the technical evaluation findings listed in NUREG-1536 Chapter 3, Article VI.

The following statements are germane to the finding of an affirmative safety evaluation for HI-STAR 100 spent fuel storage system:

- The design and structural analysis of the HI-STAR 100 System is in full compliance with the provisions of Chapter 3 of NUREG-1536. No exceptions are taken.

- The list of Regulatory Guides, Codes, and standards presented in Section 3.6 herein is in full compliance with the provisions of NUREG-1536.

- All HI-STAR 100 structures, systems, and components (SSC) that are important to safety (ITS) are identified in Table 2.2.6. Section 1.5 contains the design drawings which describe the HI-STAR 100 SSCs in complete detail. Explanatory narrations in Subsections 3.4.3, 3.4.4, and appendices to this chapter provide sufficient textual details to allow an independent evaluation of their structural adequacy.

- The requirements of 10CFR72.24 with regard to information pertinent to structural evaluation is provided in Chapters 2, 3, and 11.

- Technical Specifications pertaining to the structures of the HI-STAR 100 System have been provided in Section 12.3 herein pursuant to the requirements of 10CFR72.26.

- A series of analyses to demonstrate compliance with the requirements of 10CFR72.122(b) and (c), and 10CFR72.24(c)(3) have been performed which show that SSCs designated as ITS possess an adequate margin of safety with respect to all load combinations applicable to normal, off-normal, accident, and natural phenomenon events. In particular, the following information is provided:

    i. Load combinations for the fuel basket, enclosure vessel, and the HI-STAR 100 overpack for normal, off-normal, accident, and natural phenomenon events are compiled in Tables 2.2.14, 3.1.1, and 3.1.3 through 3.1.5, respectively.

    ii. Stress limits applicable to the materials are provided in Subsection 3.3.

    iii. Stresses at various locations in the fuel basket, the enclosure vessel, and the HI-STAR 100 overpack have been computed by analysis. Descriptions of stress analyses are presented in Sections 3.4.3 and 3.4.4, which are further elaborated in a series of appendices listed at the end of this chapter.

iv. Factors of safety in the components of the HI-STAR 100 System are reported as below:

a.	Fuel basket	Tables 3.4.3, 3.4.9, and 3.4.11
b.	Enclosure vessel	Tables 3.4.4, 3.4.7, 3.4.8, 3.4.9, 3.4.11, 3.4.12, 3.4.13, 3.4.15
c.	HI-STAR 100 overpack	Tables 3.4.5, 3.4.6, 3.4.10, and 3.4.16-3.4.19
d.	Miscellaneous components	Tables 3.4.11 and 3.4.14
e.	Lifting devices	Subsection 3.4.3

- The structural design and fabrication details of the fuel baskets whose safety function in the HI-STAR 100 System is to maintain nuclear criticality safety, have been carried out to comply with the provisions of Subsection NG of the ASME Code (loc. cit.) Section III. The structural factors of safety, summarized in Tables 3.4.3 and 3.4.6 for all credible load combinations under normal, off-normal, accident, and natural phenomenon events demonstrate that the Code limits are satisfied in all cases. As the stress analyses have been performed using linear elastic methods and the computed stresses are well within the respective ASME Code limits, it follows that the physical geometry of the fuel basket will not be altered under any load combination to create a condition adverse to criticality safety. This conclusion satisfies the requirement of 10CFR72.124(a), with respect to structural margins of safety for SSCs important to nuclear criticality safety.

  Structural margins of safety during handling, packaging, and transfer operations, mandated by the provisions of 10CFR Part 72.236(b), require that the lifting and handling devices are engineered to comply with the stipulations of ANSI N14.6, NUREG-0612, Regulatory Guide 3.61, and NUREG-1536, and that the components being handled meet the applicable ASME Code service condition stress limits. The requirements of the governing codes for handling operations are summarized in Subsection 3.4.3 herein. A summary table of factors of safety for all ITS components under lifting and handling operations, presented in Subsection 3.4.3, shows that adequate structural margins exist in all cases.

- Consistent with the requirements of 10CFR72.236(i), the confinement boundary for the HI-STAR 100 System has been engineered to maintain confinement of radioactive materials under normal, off-normal, and postulated accident conditions. This assertion of confinement integrity is made principally on the strength of the following information provided in this FSAR.

  i. The MPC Enclosure Vessel which constitutes the confinement boundary is designed and fabricated in accordance with Section III, Subsection NB (Class 1 nuclear components) of the ASME Code to the maximum extent practicable.

ii. The MPC lid of the MPC Enclosure Vessel is welded using a strength groove weld and is subjected to volumetric examination, hydrostatic testing, liquid penetrant (root and final), and leakage testing to establish maximum confidence in weld joint integrity.

iii. The closure of the MPC Enclosure Vessel consists of *two* independent isolation barriers.

iv. The confinement boundary is constructed from stainless steel alloys with a proven history of material integrity under environmental conditions encountered in terrestrial applications.

v. The load combinations for normal, off-normal, accident, and natural phenomena events have been compiled (Table 2.2.14) and applied on the HI-STAR 100 MPC Enclosure Vessel (confinement boundary) and on the HI-STAR 100 overpack. The results summarized in Tables 3.4.4 through 3.4.19, show that the factor of safety (with respect to the appropriate ASME Code limits) is greater than one in all cases. Design Basis natural phenomena events such as tornado-borne missiles (large, intermediate, or small) have also been analyzed to evaluate their potential for breaching the helium retention boundary and the confinement boundary. Analyses presented in Subsection 3.4.8 (summarized in unnumbered tables in Subsection 3.4.8 and in the appendices to this chapter), show that the integrity of the helium retention boundary and the confinement boundary is preserved under all design basis projectile impact scenarios.

- The information on structural design included in this FSAR complies with the requirements of 10CFR72.120 and 10CFR72.122, and can be ascertained from the information contained in Table 3.7.1.

- The provisions of features in the HI-STAR 100 structural design, listed in Table 3.7.2, demonstrate compliance with the specific requirements of 10CFR72.236(e), (f), (g), (h), (i), (j), (k), and (m).

Table 3.7.1 (continued)

COMPLIANCE OF HI-STAR 100 SYSTEM WITH 10CFR72.236(e), ET AL.

Item	Compliance	Location of Supporting Information in This Document
i. Design and fabrication to acceptable quality standards	All ITS components designed and fabricated to recognized Codes and Standards: • Basket: Subsection NG, Section III • Enclosure Vessel: Subsection NB, loc. cit. • HI-STAR 100 Structure: Subsection NF, loc. cit.	Subsections 2.0.1 and 3.1.1 Tables 2.2.6 and 2.2.7 Subsections 2.0.1 and 3.1.1 Tables 2.2.6 and 2.2.7 Subsections 2.0.2 and 3.1.1
ii. Erection to acceptable quality standards	HI-STAR 100 will be installed in a vertical orientation using proven deployment procedures which are in full compliance with established construction practices at nuclear power plants.	Section 8.1
iii. Testing to acceptable quality standards	• All non-destructive examination of ASME Code components for provisions in the Code (see exceptions in Table 2.2.15). • Hydrotest of pressure vessel per the Code (see Table 12.3.18). • Testing for radiation containment per provisions of NUREG-1536 (see Tables 12.3.8 and 12.3.9).	Section 9.1 Section 9.1 Sections 7.1 and 9.1
iv. Adequate structural protection against environmental conditions and natural phenomena.	Analyses presented in Chapter 3 demonstrate that the confinement boundary will preserve its integrity under all postulated off-normal and natural phenomena events listed in Chapter 2.	Section 2.2 Chapter 11

COMPLIANCE OF HI-STAR 100 SYSTEM WITH 10CFR72.236(e), ET AL.

Item	Compliance	Location of Supporting Information in This Document
v. Adequate protection against fires and explosions	• The extent of combustible (exothermic) material in the vicinity of the cask system is procedurally controlled (the sole source of hydrocarbon energy is diesel in the tow vehicle).	Chapter 8
	• Analyses show that the heat energy released from the postulated fire accident condition surrounding the cask will not result in impairment of the confinement boundary and will not lead to structural failure of the overpack. The effect on shielding will be localized to the external surfaces directly exposed to the fire which will cause no significant change in the HI-STAR 100 overpack.	Subsection 11.2.4
	• Explosion effects are shown to be bounded by the Code external pressure design basis. Pressure pulse from explosion will act on the HI-STAR 100 overpack; the MPC (confinement boundary) is completely protected.	Subsection 11.2.11 and Subsection 3.1.2.1.1.4
vi. Appropriate inspection, maintenance, and testing	Inspection, maintenance, and testing requirements set forth in this FSAR are in full compliance with the governing regulations and established industry practice.	Sections 9.1 and 9.2 Chapter 12
vii. Adequate accessibility in emergencies	The HI-STAR 100 overpack lid can be removed to gain access to the multi-purpose canister.	Chapter 8

Table 3.7.1 (continued)

COMPLIANCE OF HI-STAR 100 SYSTEM WITH 10CFR72.236(e), ET AL.

Item	Compliance	Location of Supporting Information in This Document
viii. A confinement barrier that acceptably protects the spent fuel cladding during storage	• The peak temperature of the fuel cladding at design basis heat duty of each MPC has been demonstrated to be maintained below the limits recommended in the reports of national laboratories. • The confinement barriers consist of highly ductile stainless steel alloys. The multi-purpose canister is housed in the overpack, built from a steel structure whose materials are selected and examined to maintain protection against brittle fracture under off-normal ambient (cold) temperatures (minimum of -40ºF).	Subsection 4.4.2 Subsection 3.1.1 Subsection 3.1.2.3
ix. The structures are compatible with the appropriate monitoring systems.	The HI-STAR 100 overpack	Section 1.5 Subsection 2.3.3.2
x. Structural designs that are compatible with ready retrievability of fuel.	• The fuel basket is designed to be an extremely stiff honeycomb structure such that the storage cavity dimensions will remain unchanged under all postulated normal and accident events. Therefore, the retrievability of the spent nuclear fuel from the basket will not be jeopardized. • The MPC canister lid is attached to the shell with a groove weld which is made using an automated welding device. A similar device is available to remove the weld. Thus, access to the fuel basket can be realized.	Subsection 3.1.1 Sections 8.1 and 8.3

COMPLIANCE OF HI-STAR 100 SYSTEM WITH 10CFR72.236(e), ET AL.

Item		Compliance	Location of Supporting Information in This Document
i.	Redundant sealing of confinement systems	Two physically independent lids, each separately welded to the MPC shell (Enclosure Vessel shell) provide a redundant confinement system.	Section 1.5 Drawing Nos. 1392, 1395, and 1401; Figure 7.1.2 Section 7.1.
ii.	Adequate heat removal without active cooling systems.	Thermal analyses presented in Chapter 4 show that the HI-STAR 100 System will remove the decay heat generated from the stored spent fuel by strictly passive means and maintain the system temperature within prescribed limits.	Sections 4.4 and Sections 9.1 and 9.2
iii.	Storage of spent fuel for a minimum of 20 years.	The service life of the overpack and MPC are engineered to be in excess of 20 years.	Subsections 3.4.10 and 3.4.11
iv.	Compatibility with wet or dry spent fuel loading and unloading facilities	The system is designed to eliminate any significant material interactions in the wet (spent fuel pool) environment.	Subsection 3.4.1
v.	Ease of decontamination	The external surface of the multi-purpose canister is protected from contamination during fuel loading through a custom designed sealing device.	Figures 8.1.13 and 8.1.14
vi.	Inspections of defects that might reduce confinement effectiveness.	Post-fabrication inspection of the shielding materials will be performed to ensure that no HI-STAR 100 Systems containing unacceptable voids are deployed at the ISFSI for long-term storage.	Section 9.1 and Chapter 12
vii.	Conspicuous and durable marking.	The stainless steel lid of each MPC will have model number and serial number engraved for ready identification.	Section 1.5

Table 3.7.1 (continued)

COMPLIANCE OF HI-STAR 100 SYSTEM WITH 10CFR72.236(e), ET AL.

Item	Compliance	Location of Supporting Information in This Document
viii. Compatibility with removal of the stored fuel from the site, transportation, and ultimate disposal by the U.S. Department of Energy.	The MPC is designed to be in full compliance with the DOE's draft specification for transportability and disposal published under the now dormant "MPC" program.	Section 2.4 Subsection 1.2.1.1

3.8 REFERENCES

[3.0.1] HI-STAR 100 SAR, Holtec Report HI-951251, Revision 15, Chapter 2, Docket No. 71-9261.

[3.0.2] HI-STORM FW FSAR, Holtec Report HI-2084239, Revision 1, Chapter 3, Docket No. 72-1032.

[3.1.1] NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", United States Nuclear Regulatory Commission.

[3.1.2] ANSI N14.6-1993, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500 kg) or More for Nuclear Materials", American National Standards Institute, Inc.

[3.1.3] Regulatory Guide 7.11, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches".

[3.1.4] Regulatory Guide 7.12, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater Than 4 Inches but Not Exceeding 12 Inches".

[3.1.5] NUREG/CR-1815, "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick".

[3.1.6] Aerospace Structural Metals Handbook, Manson.

[3.1.7] Armco Product Data Bulletin S-22.

[3.1.8] Regulatory Guide 7.8, "Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material", Revision 1, March, 1989, U.S. Nuclear Regulatory Commission.

[3.1.9] NUREG/CR-3826, "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Greater than Four Inches Thick."

[3.3.1] ASME Boiler & Pressure Vessel Code, Section II, Part D, 1995.

[3.4.1] ANSYS 5.2, ANSYS, Inc., 1995.

Table 3.7.1 (continued)

COMPLIANCE OF HI-STAR 100 SYSTEM WITH 10CFR72.236(e), ET AL.

[3.4.2] ASME Boiler & Pressure Vessel Code, Section III, Subsection NF, 1995.

[3.4.3] ASME Boiler & Pressure Vessel Code, Section III, Appendices, 1995.

[3.4.4] ASME Boiler & Pressure Vessel Code, Section III, Subsection NB, 1995.

[3.4.5] Mok, Fischer, Hsu, "Stress Analysis of Closure Bolts for Shipping Casks" (NUREG/CR 6007 UCRL-ID-110637), Lawrence Livermore National Laboratory/Kaiser Engineering, 1993.

[3.4.6] Code Case N-284, "Metal Containment Shell Buckling Design Methods", Section III, Division 1, Class MC, Approval Date 8/25/80.

[3.4.7] NRC Bulletin 96-04: Chemical, Galvanic or Other Reactions in Spent Fuel Storage and Transportation Casks, July 5, 1996.

[3.4.8] Theory of Elastic Stability, S.P. Timoshenko and J. Gere, McGraw Hill, 2nd Edition.

[3.4.9] Marks Standard Handbook for Mechanical Engineering, 9th ed.

[3.4.10] ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, 1995.

[3.4.11] 10CFR71, Waste Confidence Decision Review, USNRC, September 11, 1990.

[3.4.12] NUREG/CR-6322, Buckling Analysis of Spent Fuel Basket, Lawrence Livermore National Laboratory, May, 1995.

[3.4.13] Holtec report HI-2012786, Structural Calculation Package for HI-STAR Overpack, Revision 2.

[3.4.14] Holtec report HI-2012787, Structural Calculation Package for MPC, Revision 16.

[3.5.1] Chun, Witte, Schwartz, "Dynamic Impact Effects on Spent Fuel Assemblies", UCID-21246, Lawrence Livermore National Laboratory, October 20, 1987.

APPENDIX 3.A: HI-STAR DECELERATION UNDER POSTULATED DROP EVENTS AND TIPOVER

3.A.1 INTRODUCTION

Handling accidents with a HI-STAR overpack containing a loaded MPC are credible events (Section 2.2.3). The stress analyses carried out in Chapter 3 of this safety analysis report assume that the inertial loading on the load bearing members of the MPC, fuel basket, and the overpack due to a handling accident are limited by the Table 3.1.2 decelerations. The maximum deceleration experienced by a structural component is the product of the rigid body deceleration sustained by the structure and the dynamic load factor (DLF) applicable to that structural component. The dynamic load factor (DLF) is a function of the contact impulse and the structural characteristics of the component. A solution for dynamic load factors is provided in Supplement 25 of [3.4.13].

The rigid body deceleration is a strong function of the load-deformation characteristics of the impact interface, weight of the cask, and the drop height. For the HI-STAR 100 System, the weight of the structure and its surface compliance characteristics are known. However, the contact stiffness of the ISFSI pad (and other surfaces over which the HI-STAR 100 may be carried during its movement to the ISFSI) is site-dependent. The contact resistance of the collision interface, which is composed of the HI-STAR 100 and the impacted surface compliances, therefore, is not known a priori for a site. For conservatism, the HI-STAR 100 cask is simulated as a rigid body (infinite surface stiffness) which has the effect of maximizing the stiffness of the contact interface. Analyses for the rigid body decelerations are presented here for a reference ISFSI pad (which is the pad used in a recent Lawrence Livermore National Laboratory report). The surface compliance of the pad, therefore, is the only source of interface deformation in the dynamic simulations considered in this appendix.

An in-depth investigation by the Lawrence Livermore Laboratory (LLNL) into the mechanics of impact between a cask-like impactor on a reinforced concrete slab founded on a soil-like subgrade has identified three key parameters, namely, the thickness of the concrete slab, t_p, compressive strength of the concrete, f_c’ and equivalent Young’s Modulus of the subgrade, E. These three parameters are key variables in establishing the stiffness of the pad under impact scenarios. The LLNL reference pad parameters, which we hereafter denote as Set A, provide one set of values of t_p, f_c’, and E which are found to satisfy the deceleration criteria applicable to the HI-STAR 100 cask. Another set of parameters, referred to as Set B herein, are also shown to satisfy the g-load limit requirements. In fact, an infinite number of combinations of t_p, f_c’, and E can be compiled which would meet the g-load limit qualification. However, in addition to satisfying the g-limit criterion, the pad must be demonstrated to possess sufficient flexural and shear stiffness to meet the ACI 318 strength limits under factored load combinations. The minimum strength requirement to comply with ACI 318 provisions places a restriction on the lower bound values of t_p, f_c’, and E which must be met in an ISFSI pad design.

Our focus in this appendix, however, is to quantify the peak decelerations that would be experienced by a loaded HI-STAR 100 cask under the postulated impact scenarios for two pad designs defined by parameter Sets A and B, respectively. The information presented in this appendix also serves to further authenticate the veracity of the Holtec DYNA3D model described in the 1997 benchmark report [3.A.4.].

3.A.2 Purpose

The purpose of this appendix is to demonstrate that the rigid body decelerations are sufficiently low so that the design basis deceleration of 60g is not exceeded. Three scenarios of accidental drop of a loaded HI-STAR 100 cask on the ISFSI pad are considered in this appendix. They are:

i. Side drop: A loaded HI-STAR 100 free-falls in a horizontal orientation (cask's axis is horizontal) from a height "h" before impacting the ISFSI pad.

ii. Tipover: A loaded HI-STAR 100 is assumed to undergo a non-mechanistic tipover event at an ISFSI pad resulting in an impact with a pre-incipient impact angular velocity of ω which is readily calculated from elementary dynamics.

iii. End drop: The loaded cask is assumed to drop with its longitudinal axis in the vertical orientation such that its bottom plate hits the pad after free-falling from a height, h.

It is shown in Supplement 25 of [3.4.13] that dynamic load factors are a function of the dominant natural frequency of vibration of the component for a given input load pulse shape. Therefore, for the purposes of this Appendix 3.A, it is desired to demonstrate that the rigid body deceleration experienced in each of the drop scenarios is below the 60g HI-STAR 100 design basis.

3.A.3 Background and Methodology

An earlier TSAR revision of this FSAR contained an analytical treatment of the three cask drop scenarios. In the earlier submittal, the cask/ISFSI interface was simulated by a linear spring with the spring stiffeners calculated using the Bousinesq elastic half-space solution. All three scenarios reduced to the solution of a simple mass-spring system. The need for such an idealized solution was eliminated when the Lawrence Livermore National Laboratory (LLNL) published results of the so-called fourth series billet tests [3.A.1] with a companion report [3.A.2] documenting a numerical solution based methodology which simulated the drop test results with reasonable accuracy. Subsequently, USNRC personnel published a paper [3.A.3] affirming the NRC's endorsement of the LLNL methodology. The LLNL simulation used modeling and simulation algorithms contained within the commercial computer code DYNA3D [3.A.6].

The LLNL cask drop model is not completely set forth in the above-mentioned LLNL reports. Using the essential information provided by the LLNL [3.A.2] report, however, Holtec is able to develop a finite element model for implementation on DYNA3D which is fully consistent with LLNL's (including the use of the Butterworth filter for discerning rigid body deceleration from "noisy" impact data). The details of the DYNA3D dynamic model, henceforth referred to as the Holtec model, are contained in the proprietary benchmark report [3.A.4] wherein it is shown that the peak deceleration in every case of billet drop analyzed by LLNL is replicated within a small tolerance by the Holtec model. The case of the so-called "generic" cask for which LLNL provided predicted response under side drop and tipover events is also bounded by the Holtec model. In summary, the benchmarking effort documented in [3.A.4] is in full compliance with the guidance of the Commission [3.A.3].

Having developed and benchmarked an LLNL-consistent cask impact model, this model is applied to prognosticate the HI-STAR drop scenarios.

In the tipover scenario, the angular velocity of approach is readily calculated using planar rigid body dynamics and is used as an initial condition in the DYNA3D simulation.

For the side drop and end drop scenarios, considering the reference target (pad) elasto-plastic-damage characteristics, the object is to determine the maximum allowable drop height "h" such that the rigid body deceleration is below the design basis.

It is recognized, from the elementary analogy of the spring-mass impact, that the maximum deceleration increases monotonically as the rigidity of the cask is increased. Therefore, an upper bound on the deceleration of HI-STAR is obtained by replacing the polymeric zone in HI-STAR also by a rigid medium, making the entire cask a rigid body. Simulations for side drop and tipover conditions under the complete rigid body assumption provide an upper bound on the cask response. For the case of vertical drop, the impacting region is bottom plate forging which, without excessive conservatism, can be also modeled as a rigid body. Thus, all drop simulations presented in this appendix assume the HI-STAR 100 cask to simulate a rigid body for conservatism.

A description of the work effort and a summary of the results are presented in the following sections. In all cases, the reported decelerations are below the design basis limit of 60 g's at the top of the MPC fuel basket.

3.A.4 Assumptions and Input Data

3.A.4.1 Assumptions

The assumptions used to create the model are completely described in Reference [3.A.4] and are shown there to be consistent with the LLNL simulation. There are two key aspects which are restated here:

The maximum deceleration experienced by the cask during a collision event is a direct function of the structural rigidity (or conversely, compliance) of the impact surface. The compliance of the ISFSI pad is quite obviously dependent on the thickness of the pad, t_p, the compressive strength of the concrete, f_c' and stiffness of the subgrade (expressed by its effective Young's modulus, E). The structural rigidity of the ISFSI pad will increase if any of the three above-mentioned parameters, t_p, f_c' or E is increased. For the reference pad, the governing parameters (i.e., t_p, f_c' and E) are assumed to be identical to the pad defined by LLNL [3.A.2] which is also the same as the pad utilized in the benchmark report [3.A.4]. We refer to the LLNL ISFSI pad parameters as Set A. (Table 3.A.1).

As can be seen from Table 3.A.1, the nominal compressive strength f_c' in Set A is limited to 4200 psi (28.96 MPa). However, experience has shown that ISFSI owners have considerable practical difficulty in limiting the 28 day strength of poured concrete to 4200 psi (28.96 MPa), chiefly because a principal element of progress in reinforced concrete materials technology has been in realizing ever increasing concrete nominal strength. Inasmuch as a key objective of the ISFSI pad is to limit its structural rigidity (and not f_c' per se), and limiting f_c' to 4200 psi (28.96 MPa) may be problematic in certain cases, an alternative set of reference pad parameters is defined (Set B in Table 3.A.1) which permits a higher value of f_c' but much smaller values of pad thickness, t_p and sub-grade Young's modulus, E.

The ISFSI owner has the option of constructing the pad to comply with the limits of Set A or Set B without performing site-specific cask impact analyses. It is recognized that, for a specific ISFSI site, the reinforced concrete, as well as the underlying engineered fill properties, may be different at different locations on the pad or may be uniform, but non-compliant with either Set A or Set B. In that case, the site-specific conditions must be performed to demonstrate compliance with the design limits of the HI-STAR system (e.g., maximum rigid body g-load less than or equal to 60g's). The essential data which define the pad (Set A and Set B) used to qualify the HI-STAR 100 are provided in Table 3.A.1.

As the results presented in this appendix show, Set B parameters lead to a lower g-loads than the LLNL Set (Set A).

3.A.4.2 Input Data

Table 3.A.1 characterizes the properties of the reference target pad (Set A and Set B) used in the analysis.

The principal strength parameters that define the stiffness of the pad, namely t_p, E and f_c' are input in the manner described in [3.A.2] and [3.A.4].

Table 3.A.2 details the geometry of the HI-STAR 100 used in the drop simulations. This data is taken from applicable HI-STAR 100 drawings.

3.A.5 Finite Element Model

The finite-element model of the Holtec HI-STAR 100 cask (bottom plate, shells, forging, lid, Holtite polymer and its connectors), concrete pad and a portion of the subgrade soil is constructed using the pre-processor integrated with the LS-DYNA3D software [3.A.5]. The deformation field for all postulated drop events, the end-drop, the side-drop and the tipover, exhibits symmetry with the vertical plane passing through the vertical diameter of the cask and the concrete pad length. Using this symmetry condition of the deformation field a half finite-element model is constructed. The finite-element model is organized into five independent parts (the cask, the MPC steel plates, the basket fuel zone, the concrete pad and the soil). The final model contains 35431 nodes, 29944 solid type finite-elements, five (5) materials, one (1) property and four (4) interfaces. The finite-element model used for the side-drop and tipover-drop events is depicted in Figures 3.A.1 through 3.A.4. Figures 3.A.5 through 3.A.8 show the end-drop finite-element model.

The half portion of the cylindrical cask contains 7,320 solid finite-elements. Figure 3.A.11 depicts details of the cask finite-element mesh.

The elasto-plastic behavior characteristics of all HI-STAR components (shells, lids, Holtite, outer skin, connectors, etc.) are simulated as rigid materials using a DYNA3D built-in command.

The soil grid, shown in Figure 3.A.9, is a rectangular prism (800 inches [20.32 m] long, 375 inches [9.52 m] wide and 470 inches [11.94 m] deep), is constructed from 13294 solid type finite-elements. The material defining this part is an elastic orthotropic material. The central portion of the soil (400 inches [10.16 m] long, 150 inches [3.81 m] wide and 170 inches [4.32 m] deep) where the stress concentration is expected to appear is discretized with a finer mesh.

The concrete pad is 320 inches [8.128 m] long, 100 inches [2.54 m] wide and is 36 inches [0.91 m] thick. This part contains 8208 solid finite-elements. A uniform sized finite-element mesh, shown in Figure 3.A.10, is used to model the concrete pad. The concrete behavior is described using a special constitutive law and yielding surface contained within DYNA3D. The geometry, the material properties, and the material behavior are identical to the LLNL reference pad.

The MPC and the contained fuel is modeled in two parts which represent the lid and baseplate, and the fuel area. An elastic material is used for both parts. The finite-element mesh pertinent to the MPC contains 1122 solid finite-elements and is shown in Figure 3.A.14. The mass density is appropriate to match a representative weight of 241,937 lb (1,076 kN) which is an approximate mean of the upper and lower weight estimates for a loaded HI-STAR 100. The total weight used in the analysis is approximately 8,000 lb (35.58 kN) heavier than the HI-STAR 100 containing the lightest weight MPC.

Analysis of a single mass impacting a spring with a given initial velocity shows that both the maximum deceleration "a_M" of the mass and the time duration of contact with the spring "t_c" are related to the dropped weight "w" and drop height "h" as follows:

$$a_M \sim \frac{\sqrt{h}}{\sqrt{w}}; t_c \sim \sqrt{w}$$

Therefore, the most conservatism is introduced into the results by using the minimum weight. However, since the difference between the heaviest and the lightest HI-STAR 100 is only 9,500 lb (42.26 kN), a small percentage of the total weight, the results using the minimum weight will yield a 2% increase in the maximum deceleration and a 2% decrease in the duration of the impact. This small difference is neglected in the presentation of results.

It is emphasized that the finite element model described in the foregoing is identical in its approach to the "Holtec model" described in the benchmark report [3.A.4]. Gaps between the MPC and the overpack are included in the model.

3.A.6 Impact Velocity

a. Linear Velocity: Vertical Drops

For the side drop and vertical drop events, the impact velocity, υ, is readily calculated from the Newtonian formula:

$$\upsilon = \sqrt{(2gh)}$$

where

g = acceleration due to gravity
h = free-fall height

b. Angular velocity: Tipover

The tipover event is an artificial construct wherein the HI-STAR 100 overpack is assumed to be perched on its edge with its C.G. directly over the pivot point A (Figure 3.A.15). In this orientation, the overpack begins its downward rotation with zero initial velocity. At angle ψ_1 (Figure 3.A.17), the pivot point shifts to point B; but otherwise the downward rotation of the overpack continues with increasing angular velocity. Towards the end of the tipover, the overpack is horizontal with its downward velocity ranging from zero at the pivot point to a maximum at the farthest point of impact (point E in Figure 3.A.18). The angular velocity at the instant of impact defines the downward velocity distribution along the contact line.

In the following, we derive an explicit expression for calculating the angular velocity of the cask at the instant when it impacts on the ISFSI pad.

Referring to Figure 3.A.15, let r be the length AC where C is the cask centroid. Therefore,

$$r = \left(\frac{d^2}{4} + (h + a)^2 \right)^{1/2} \qquad \textbf{(3.A.1)}$$

The mass moment of inertia of the HI-STAR 100 System, considered as a rigid body, can be written about an axis through point A, as

$$I_A = I_c + \frac{W}{g} r^2 \quad \textbf{(3.A.2)}$$

where I_c is the mass moment of inertia about a parallel axis through the cask centroid C, and W is the weight of the cask (W = Mg).

Let $\theta_1(t)$ be the rotation angle between a vertical line and the line AC. The equation of motion for rotation of the cask around point A, during the time interval prior to contact with point B (Figure 3.A.15), is

$$I_A \frac{d^2\theta_1}{dt^2} = \text{Mgr} \sin \theta_1 \quad \textbf{(3.A.3)}$$

This equation can be rewritten in the form

$$\frac{I_A}{2} \frac{d\left(\dot{\theta}_1\right)^2}{d\theta_1} = \text{Mgr} \sin \theta_1 \quad \textbf{(3.A.4)}$$

which can be integrated over the limits $\theta_1 = 0$ to $\theta_1 = \psi_1$. (See Figure 3.A.17).

The final angular velocity $(d\theta_1/dt)_B$ at the time instant just prior to contact at point B is given by the expression

$$\dot{\theta}_1(t_B) = \sqrt{\frac{2Mgr}{I_A}(1 - \cos\psi_1)} \quad \textbf{(3.A.5)}$$

The angle ψ_1 between AC and the vertical, at the time just prior to contact, is given by geometry as

$$\psi_1 = \psi_A - \psi_B \quad \textbf{(3.A.6)}$$

where

$$\psi_A = \tan^{-1}\left(\frac{a}{b}\right)$$

$$\psi_B = \tan^{-1}\left(\frac{d}{2h}\right)$$

and a and b are shown in Figure 3.A.17. At contact with point B at time t_B (Figure 3.A.16), the angular impulse momentum equation can be used to determine a new initial angular velocity for subsequent determination of the angular motion about point B. Ignoring the small impulsive moment from the cask weight due to the instantaneous change in moment arm, the angular momentum balance gives

$$I_A \dot{\theta}_1(t_B) = I_B \dot{\theta}_2(0) \tag{3.A.7}$$

where $I_B = I_C + Mr_1^2$ is the mass moment of inertia of the cask about point B. Solving for $\dot{\theta}_2(0)$ and eliminating $\dot{\theta}_1(t_b)$ using Eq. (3.A.5) gives

$$\dot{\theta}_2(0) = \sqrt{\frac{2Mgr_1}{I_B} \frac{I_A}{I_B} (1 - \cos\psi_1) \left(\frac{r}{r_1} \right)} \tag{3.A.8}$$

The angle $\theta_2(0)$, which is the starting point for the rotational motion around point B, is easily obtained from the cask geometry. With X defined in Figure 3.A.16.

$$\sin\theta_2(0) = \frac{X(0)}{r_1} \tag{3.A.9}$$

where X(0) can be determined from Figure 3.A.17 as

$$X(0) = r\sin\psi_1 - \left(a^2 + b^2\right)^{1/2}$$

so that

$$\sin\theta_2(0) = \frac{r}{r_1}\sin\psi_1 - \frac{\left(a^2 + b^2\right)^{1/2}}{r_1} \tag{3.A.10}$$

where

$$r_1 = \left[\left(\frac{D}{2} \right)^2 + h^2 \right]^{1/2}$$

With the initial conditions determined by Eq. (3.A.9) and (3.A.10), the solution for the motion is easily obtained.

The angular velocity $(d\theta_2/dt)_f$ at the instant of ground contact is

$$\dot{\theta}_{2f} - \dot{\theta}_2(0) = \frac{2Mgr_1}{I_B}\left(\cos\theta_2(0) - \cos\theta_{2f}\right) \quad \textbf{(3.A.11)}$$

where, from Figure 3.A.18

$$\theta_{2f} = \cos^{-1}\left(\frac{D}{2r_1}\right) \quad \textbf{(3.A.12)}$$

Using Eq. (3.A.8) to eliminate $(d\theta_2/dt)_0$ from Eq. (3.A.11) leads to a solution for the angular velocity $(d\theta_2/dt)_f$ when interface contact occurs, in the form

$$\dot{\theta}_{2f} = \sqrt{\frac{2Mgr_1}{I_B}\beta} = \omega \quad \textbf{(3.A.13)}$$

where, $$\beta = \frac{I_A r}{I_B r_1}\left(1 - \cos\psi_1\right) + \cos\theta_2(0) - \cos\theta_{2f} \quad \textbf{(3.A.14)}$$

Equations (3.A.13) and (3.A.14) establish the initial conditions for the final phase of the tipover analysis; namely, the portion of the motion when the cask is decelerated by the resistive force at the ISFSI pad interface.

Using the data germane to HI-STAR 100 (Table 3.A.2), and the above equations, the angular velocity of impact is calculated as 1.79 rad/sec.

3.A.7 Results

3.A.7.1 Set A Pad Parameters:

The LS-DYNA3D time-history results are processed using the Butterworth filter (in conformance with the LLNL methodology) to establish the time-history rigid body motion of the cask. The material points on the cask where the acceleration displacement and velocity are computed for each of the three drop scenarios are shown in Figure 3.A.19.

Node 2901 (Channel A2), which is located midway on the outermost shell generator at the top in side drop events serves as the reference point.

Node 5151 (Channel A1), which is located at the center of the outer surface of the bottom forging, serves as the reference point for end-drop scenarios.

Node 6000 (Channel A3), which is located at the center of the cask top lid outer surface, serves as the reference point for the tipover scenario with the pivot point indicated as Point 0 in Figure 3.A.19.

The results reported below for maximum cask-ISFSI contact force have been multiplied by 2.0 to reflect the fact that only 50% of the dropped mass is included in the model due to the symmetry assumption.

i. Side Drop:

The time-histories of the impact force and displacement, velocity, and deceleration at the reference node point (Channel A2) have been determined for a drop height, h, of 72" (1.83 m). The peak cask/pad impact force is 9.636E+06 lbs (42,863 kN) and the contact duration associated with the initial peak is 9.5 milli-seconds.

The maximum rigid body deceleration (filtered at 350 Hz cut-off frequency) is 49.67 g's, which is below the design basis limit of 60 g's. The time duration of the peak deceleration pulse is 4.4 milli-seconds.

ii. Tipover:

The time-histories of the impact force and displacement, velocity and vertical deceleration of Channel A3 (in Figure 3.A.19) for this event have been determined [3.A.7].

The deceleration at the tip of the fuel basket is obtained by ratioing the filtered deceleration of Node 6000. The maximum filtered deceleration at the tip of the fuel basket is 66.02 x 0.906 = 59.81 g's which is below the design basis limit. The 0.906 multiplier is based on the geometry of the loaded HI-STAR 100 (further explained in Table 3.A.3). The maximum contact force in this event is 6.43E+06 lbs (28,602 kN) and the contact duration associated with the initial peak is approximately 8.8 milli-seconds. It should be emphasized that the calculated deceleration for Node 6000 was filtered at 350 Hz cut-off frequency.

The duration of the initial deceleration pulse is 4.4 milli-seconds.

iii. End Drop:

As in all other impact cases analyzed in this appendix, the overpack is treated as a completely rigid body in the end drop scenario. One drop height is considered: h = 21" (0.53 m). The results are summarized in Table 3.A.3 and the contact force, displacement, velocity, and acceleration time-histories at Channel A1 (Figure 3.A.19) for the 21" (0.53 m) end drop are documented in the calculation package [3.A.7]. The duration of the contact force initial pulse is approximately 2.7 milli-seconds, and the filtered cask deceleration pulse is 2.1 milli-seconds.

A carry height of 21" (0.53 m) gives peak filtered deceleration in the event of an end drop of approximately 53 g's.

Decelerations obtained from the DYNA3D numerical solutions are filtered through a Butterworth type filter identical to the filter used by LLNL to investigate the "generic" cask [3.A.2]. The filter has the following characteristics: 350 Hz passband frequency, 10,000 Hz stopband frequency, 0.15 maximum passband ripple, and 10 minimum stopband attenuation.

The computer code utilized in this analysis is LS-DYNA3D [3.A.5] validated under Holtec's QA system. Table 3.A.3 summarizes the key results for all impact simulations for the Set A parameters discussed in the foregoing.

3.A.7.2 Set B Pad Parameters:

As stated previously, Set B parameters produce a much more compliant pad than the LLNL reference pad (Set A). This fact is borne out by the side drop, tipover and end drop analyses performed on the pad defined by the Set B parameters. Table 3.A.4 provides the filtered results for the three impact scenarios. In every case, the peak decelerations corresponding to Set B parameters are less than those for Set A (provided in Table 3.A.3).

Impact force and acceleration time history curves for Set B have the same general shape as those for Set A and are contained in the calculation package. All significant results are summarized in Table 3.A.4.

3.A.8 <u>Computer Codes and Archival Information</u>

The input and output files created to perform the analyses reported in this appendix are archived in the Holtec International calculation package.

3.A.9 <u>Conclusion</u>

The DYNA3D analysis of HI-STAR 100 reported in this appendix leads to the following conclusion:

a. If a loaded HI-STAR undergoes a free fall for a height of 21 inches (0.53 m) in a vertical orientation, the maximum rigid body deceleration is limited to 52.26 g's and 50.25 g’s for Set A and Set B pad parameters, respectively.

b. If a loaded HI-STAR 100 undergoes a free fall in a horizontal orientation (side drop) for a height 72" (1.83 m), the maximum rigid body deceleration is limited to 49.67 and 46.77g’s for Set A and Set B pad parameters, respectively.

c. If a loaded HI-STAR 100 overpack pivots about its bottom edge and tips over then the maximum rigid body deceleration of the cask centerline at the plane of the top of the fuel basket cellular region is 59.81 and 50.64 g’s respectively for pad parameter Set A and Set B.

Tables 3.A.3 and 3.A.4 provides the key results for all drop cases studied herein for pad parameter Set A and B respectively.

Recalling that the design basis g-load is 60 g's, the above impact scenarios are comfortably enveloped by the level D design limit and allow ample margin for the introduction of appropriate dynamic load factors into the component stress analyses.

If the pad designer maintains each of the three significant parameters (t_p, f_c' and E) below the limit for the specific set selected (Set A or Set B), then the stiffness of the pad at the ISFSI site will be lower and the computed decelerations at the ISFSI site will also be expected to be lower. Furthermore, because the mathematical model for the cask assumes infinite rigidity (which is not a requirement for the LLNL methodology (3.A.2) or Holtec's benchmark work effort (3.A.4), refinement of the cask dynamic model will accrue further reduction in the computed peak deceleration. Likewise, incorporation of the structure flexibility in the MPC enclosure vessel, fuel basket, etc., would lead to additional reductions in the computed values of the peak deceleration. These refinements, however, add to the computational complexity. Because g-limits are met without the above-mentioned and other refinements in the cask dynamic model, the rigid body model for HI-STAR 100 was retained to preserve simplicity.

3.A.10 References

[3.A.1] Witte, M., et al., "Evaluation of Low-Velocity Impacts Tests of Solid Steel Billet onto Concrete Pads.", Lawrence Livermore National Laboratory, UCRL-ID-126274, Livermore, California, March 1997.

[3.A.2] Witte, M., et al., "Evaluation of Low-Velocity Impacts Tests of Solid Steel Billet onto Concrete Pads, and Application to Generic ISFSI Storage Cask for Tipover and Side Drop.", Lawrence Livermore National Laboratory, UCRL-ID-126295, Livermore, California, March 1997.

[3.A.3] Tang, D.T., Raddatz, M.G., and Sturz, F.C., "NRC Staff Technical Approach for Spent Fuel Cask Drop and Tipover Accident Analysis", SFPO, USNRC (1997).

[3.A.4] Simulescu, I., "Benchmarking of the Holtec LS-DYNA3D Model for Cask Drop Events", Holtec Report HI-971779, September 1997.

[3.A.5] LS-DYNA3D, Version 936-03, Livermore Software Technology Corporation, September 1996.

[3.A.6] Whirley, R.G., "DYNA3D, A Nonlinear, Explicit, Three-Dimensional Finite element Code for Solid and Structural Mechanics - User Manual.", Lawrence Livermore National Laboratory, UCRL-MA-107254, Revision 1, 1993.

[3.A.7] Zhai, J., “Analysis of the Loaded HI-STAR 100 System Under Drop and Tip-Over Scenarios,” Holtec Report HI-2002466.

Table 3.A.1
Essential Variables to Characterize the ISFSI Pad (Set A and Set B)

Item	Parameter Set A	Parameter Set B
Thickness of concrete (inches) (meters)	36 (0.91)	28 (0.71)
Nominal compressive strength of concrete at 28 days (psi) (MPa)	4,200 (28.96)	6,000 (41.37)
Max. modulus of elasticity of the subgrade (psi) (MPa)	28,000 (193)	16,000 (110)

Notes:

1. The concrete Young's Modulus is derived from the American Concrete Institute recommended formula where f is the nominal compressive strength of the concrete (psi).

2. The effective modulus of elasticity of the subgrade shall be measured by the classical "plate test" or other appropriate means before pouring of the concrete to construct the ISFSI pad.

Table 3.A.2
Key Input Data in Drop Analyses

Cask weight	128,275 lb (570.6 kN)
Holtite weight	12,926 lb (57.5 kN)
Holtite connectors weight	11,879 lb (52.8 kN)
Length of the cask	203.125 inches (5159 mm)
Length of the Holtite	173.125 inches (4397 mm)
Diameter of the bottom plate	83.25 inches (2114 mm)
Inside diameter of the cask	68.75 inches (1746 mm)
Outside diameter of the cask shells	85.75 inches (2178 mm)
Outside diameter of the enclosure plate	96.00 inches (2438 mm)
Outside diameter of the Holtite	95.00 inches (2413 mm)
MPC weight (including fuel)	88,857 lb (395.2 kN)
MPC height	190.5 inches (4839 mm)
MPC diameter	68.375 inches (1737 mm)
MPC bottom plate thickness	2.5 inches (63.5 mm)
MPC top plate thickness	9.5 inches (241.3 mm)

Table 3.A.3
FILTERED RESULTS FOR DROP AND TIPOVER SCENARIOS (SET A)

Drop Event	Rigid Cask Model†			
	Max. Displ (in) (mm)	**Impact Velocity (in/sec) (mm/sec)**	**Max. Acc. (g's)**	**Acc. Pulse Duration (msec.)**
End-21" (533 mm)	1.144 (29.06)	127.4 (3,236)	52.26	2.1
Side-72" (1829 mm)	2.674 (67.92)	235.9 (5,992)	49.67	4.4
Tipover Top of Cask††	4.231 (107.47)	348.4 (8,849)	66.02	4.4
Tipover Top of Basket Elevation	--	--	59.81	--

† The passband frequency of the Butterworth filter is 350 Hz.

†† The distance of the top of the fuel basket is 176.25" (4,477 mm) from the pivot point. The distance of the top of the cask is 194.375" (4,937 mm) from the pivot point. Therefore, all displacements, velocities, and accelerations of the top of the fuel basket are 90.6% of the cask top (176.25/194.4) (4477/4937).

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Table 3.A.4
FILTERED RESULTS FOR DROP AND TIPOVER SCENARIOS (SET B)

	Rigid Cask Model[†]			
Drop Event	**Max. Displ (in) (mm)**	**Impact Velocity (in/sec) (mm/sec)**	**Max. Acc. (g's)**	**Acc. Pulse Duration (msec.)**
End-21" (533 mm)	1.335 (33.91)	127.4 (3,236)	50.25	2.0
Side-72" (1829 mm)	4.533 (115.14)	235.9 (5,992)	46.77	4.0
Tipover Top of Cask[††]	6.620 (168.15)	348.4 (8,849)	55.89	4.0
Tipover Top of Basket Elevation	--	--	50.64	--

[†] The passband frequency of the Butterworth filter is 350 Hz.
[††] The distance of the top of the fuel basket is 176.25" (4,477 mm) from the pivot point. The distance of the top of the cask is 194.375" (4,937 mm) from the pivot point. Therefore, all displacements, velocities, and accelerations of the top of the fuel basket are 90.6% of the cask top (176.25/194.4) (4477/4937).

“MISCELLANEOUS NUMERICAL CALCULATIONS SUPPORTING APPENDIX 3.A”
INTENTIONALLY DELETED

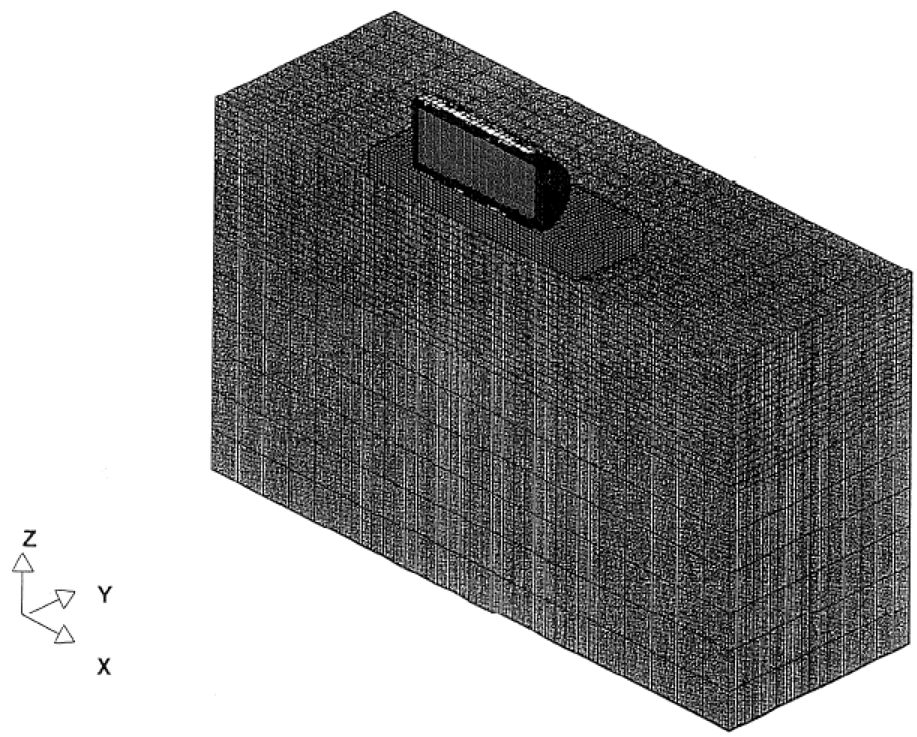

Fig. 3.A.1 Side-Drop and Tipover Finite-Element Model (3-D View)

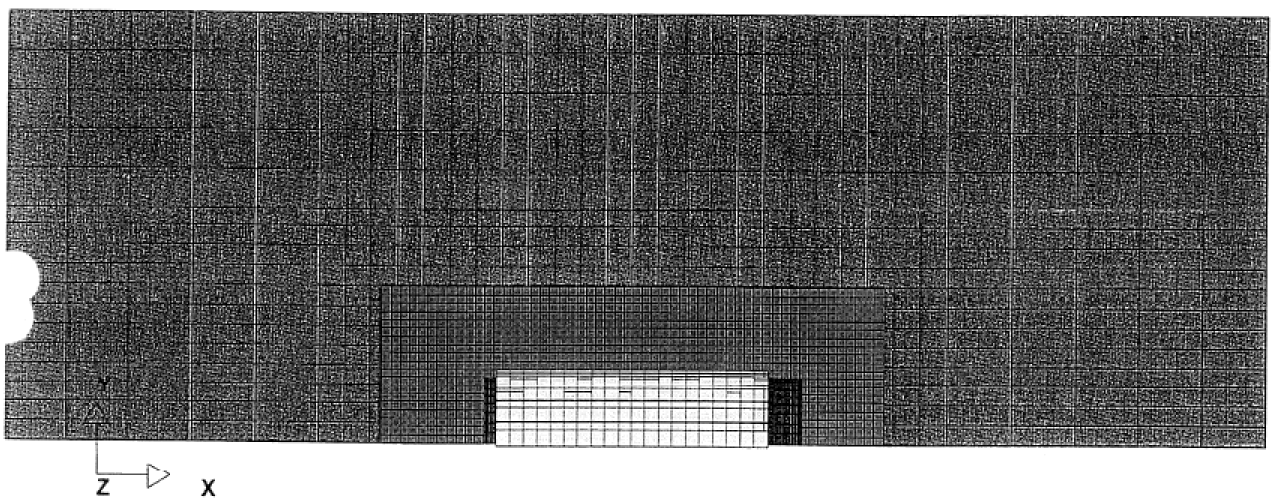

Fig. 3.A.2 Side-Drop and Tipover Finite-Element Model (Plan View)

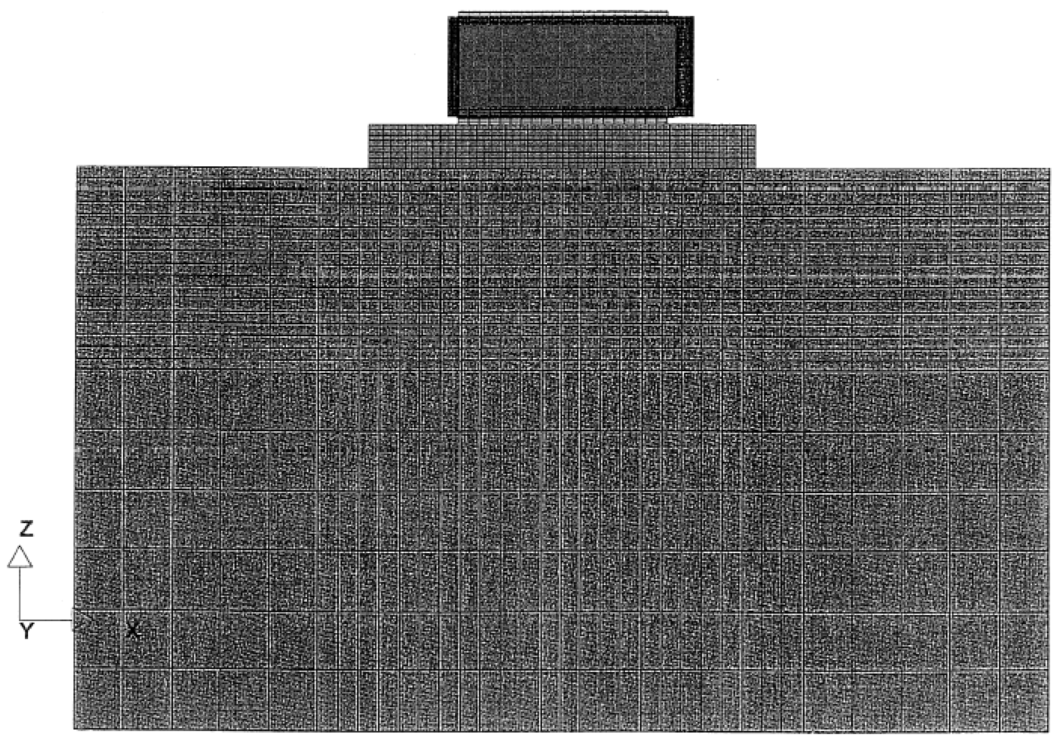

Fig. 3.A.3 Side-Drop and Tipover Finite-Element Model (XZ View)

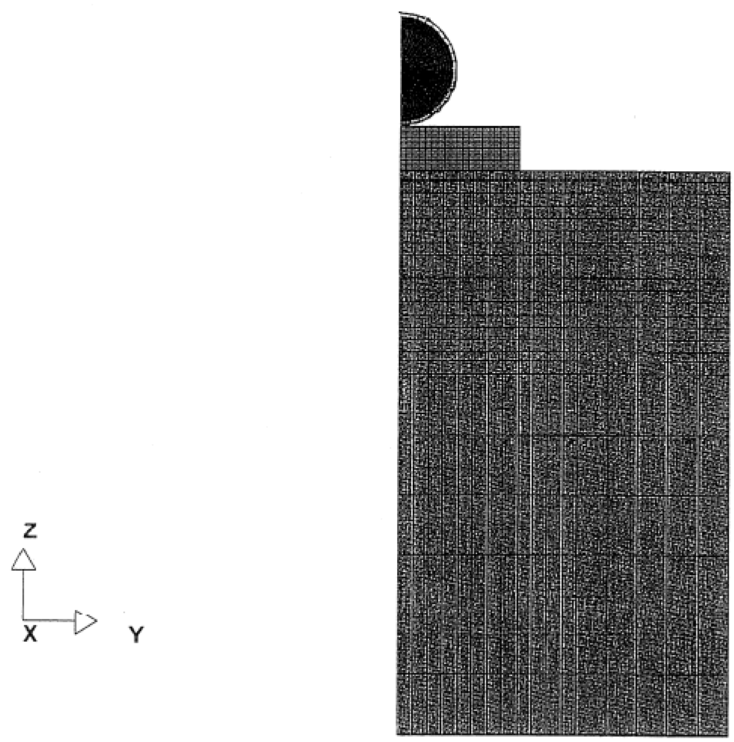

Fig. 3.A.4 Side-Drop and Tipover Finite-Element Model (YZ View)

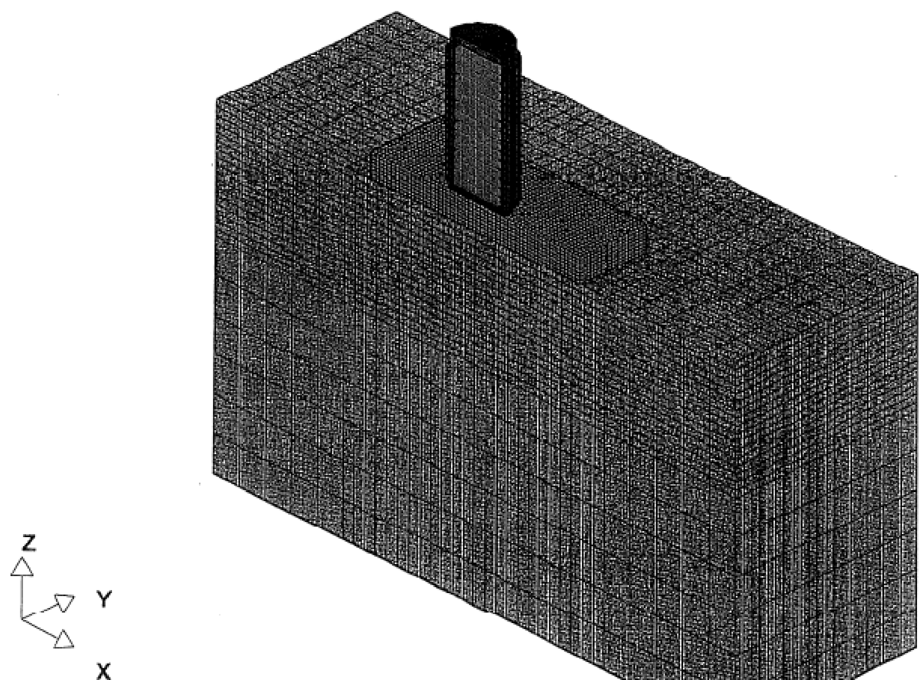

Fig. 3.A.5 End-Drop Finite-Element Model (3-D View)

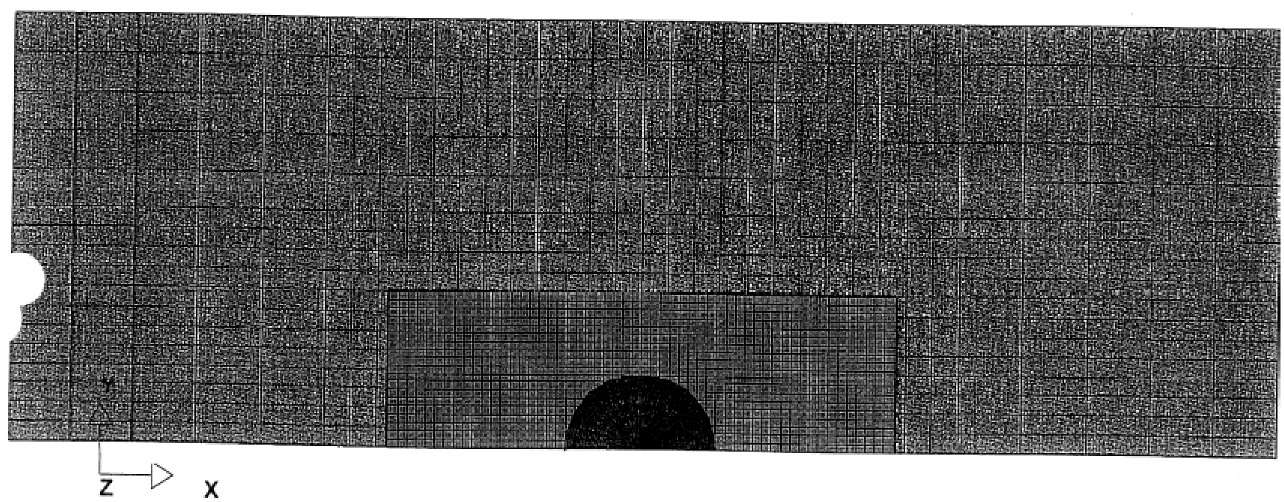

Fig. 3.A.6 End-Drop Finite-Element Model (Plan View)

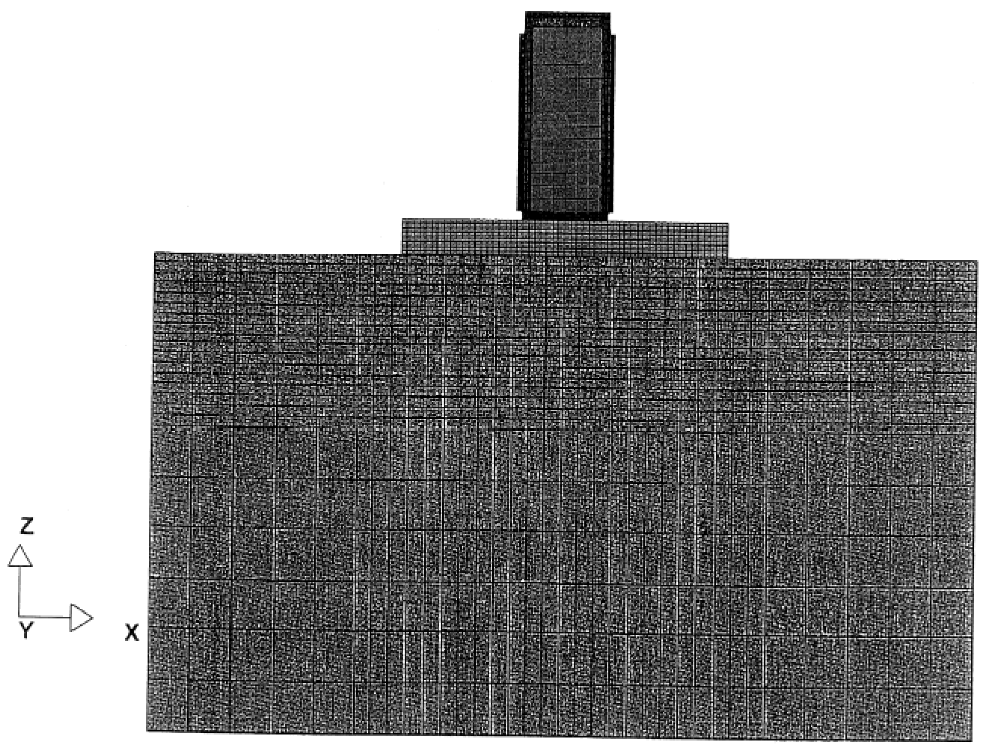

Fig. 3.A.7 End-Drop Finite-Element Model (XZ View)

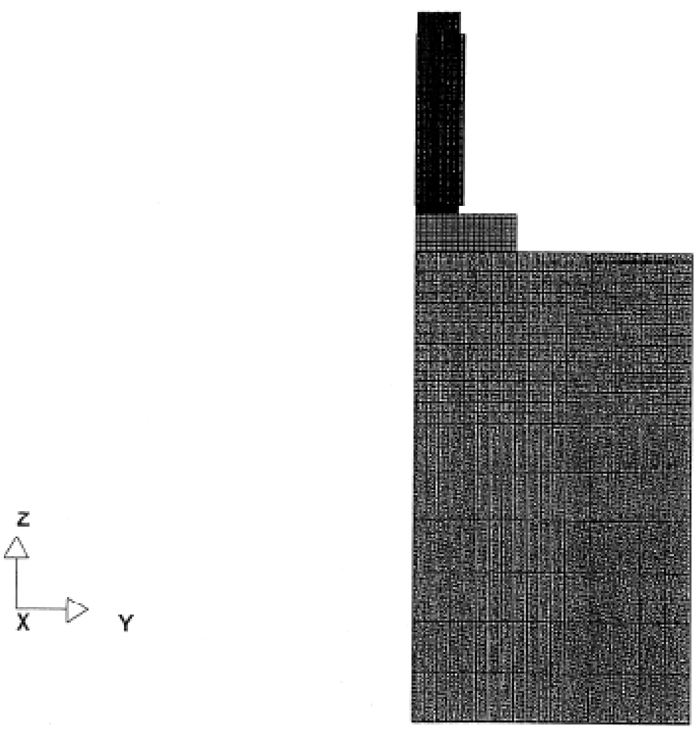

Fig. 3.A.8 End-Drop Finite-Element Model (YZ View)

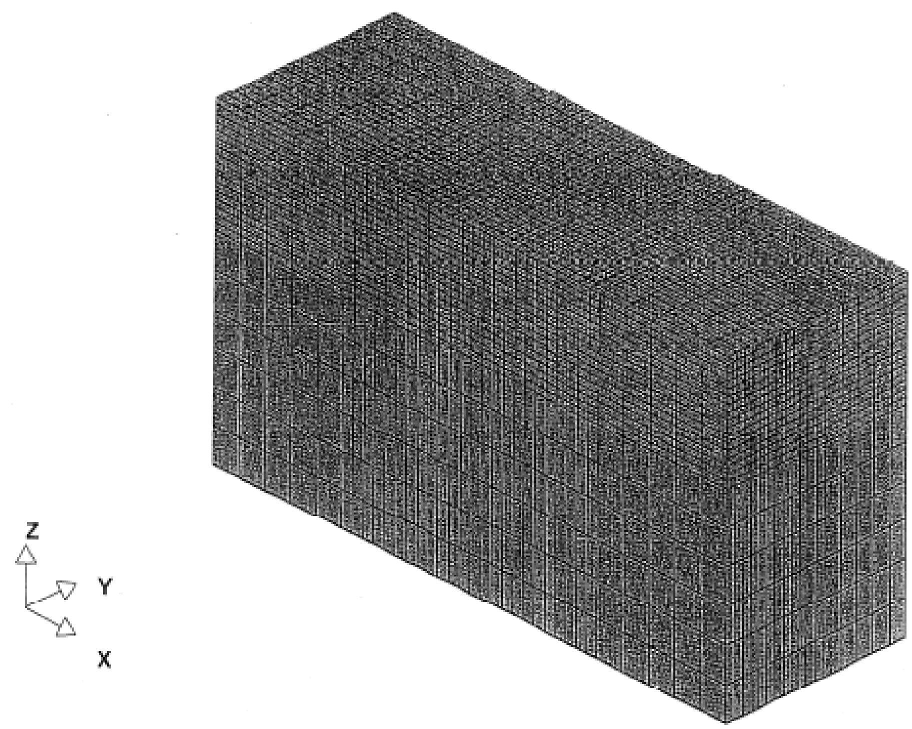

Fig. 3.A.9 Soil Finite-Element Model (3-D View)

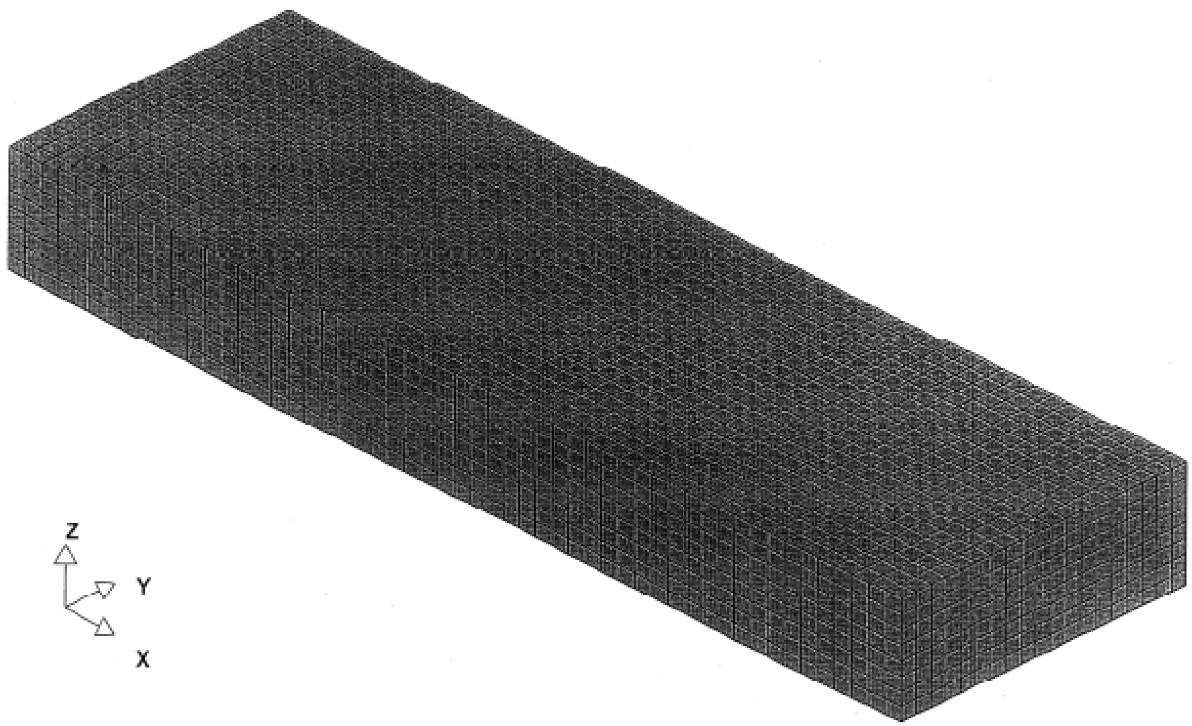

Fig. 3.A.10 Concrete Pad Finite-Element Model (3-D View)

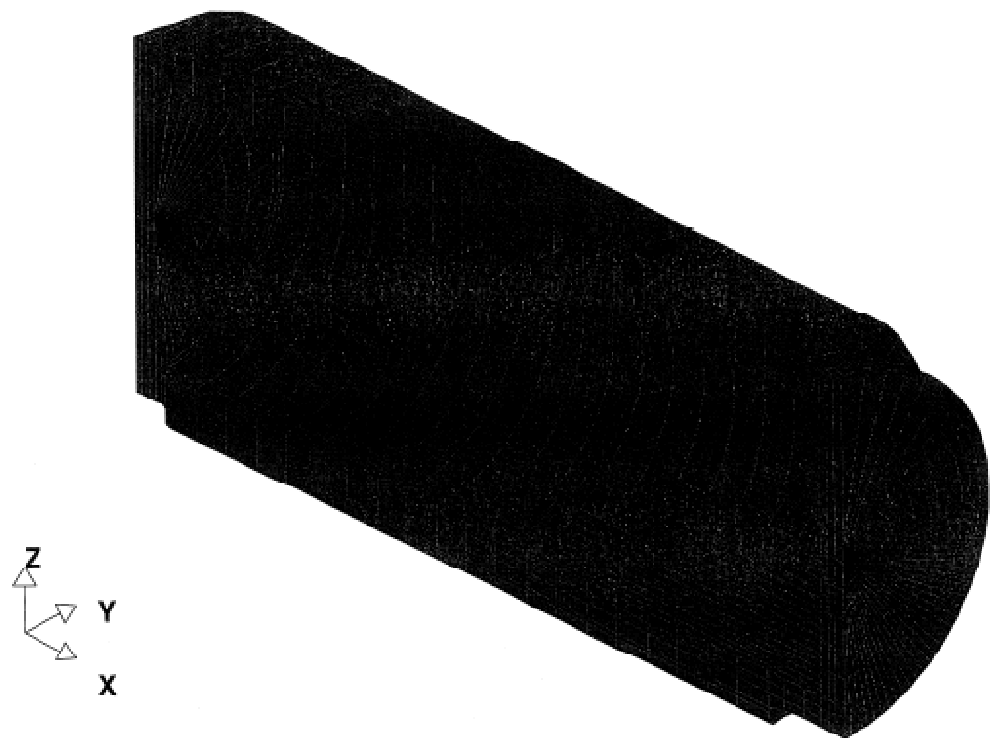

Figure 3.A.ll Cask Finite-Element Model (3-D View)

FIGURE 3.A.12 DELETED

FIGURE 3.A.13 DELETED

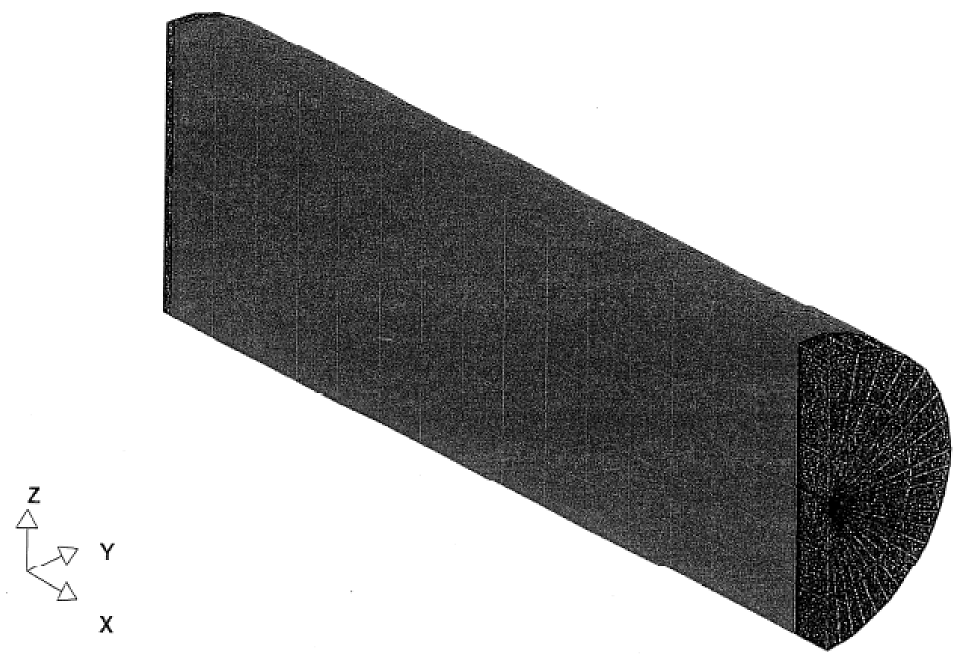

Fig. 3.A.14 MPC Finite-Element Model (3-D View)

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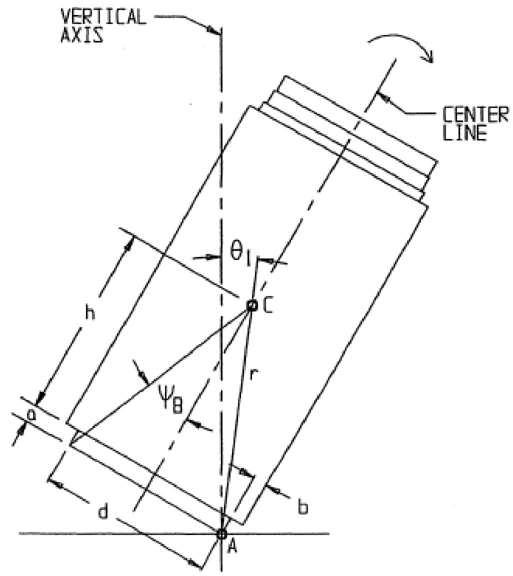

FIGURE 3.A.15; PIVOT POINT SHIFT DURING TIP-OVER
INITIAL CONDITION

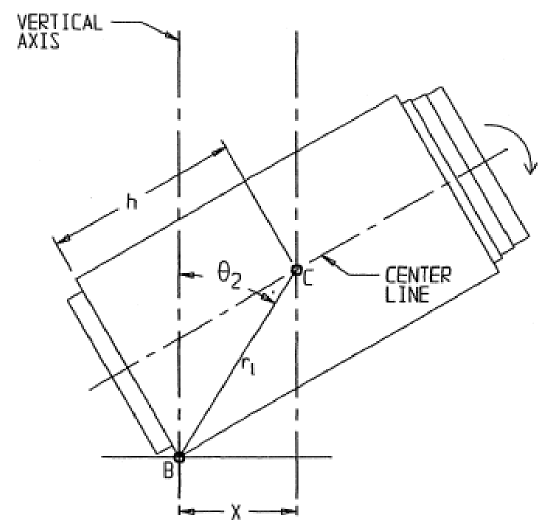

FIGURE 3.A.16; PIVOT POINT SHIFT DURING TIP-OVER
INTERMEDIATE CONDITION

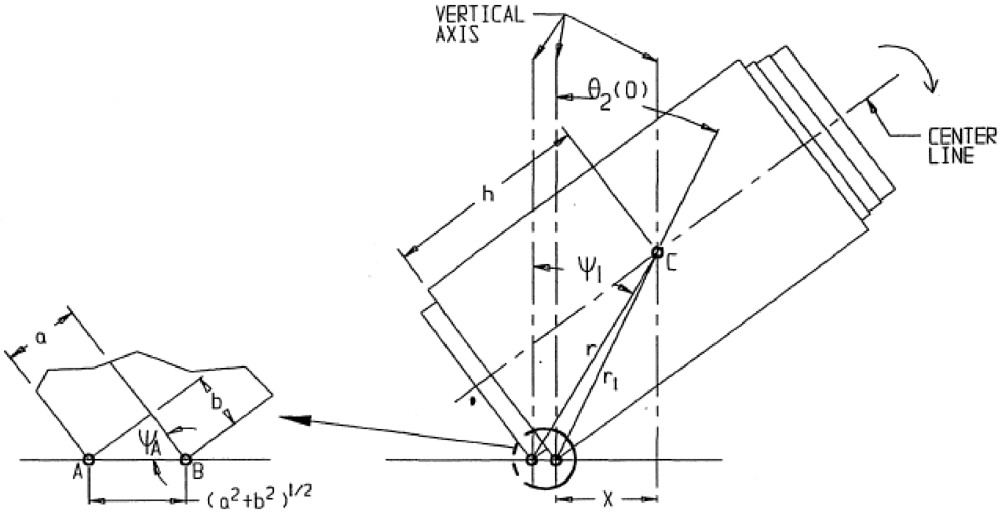

FIGURE 3.A.17; TIP-OVER EVENT AT THE INSTANT WHEN POINTS A AND B ARE BOTH IN CONTACT WITH THE GROUND

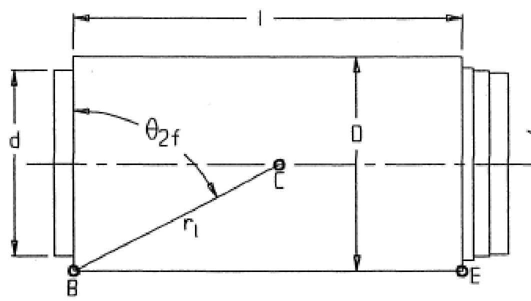

FIGURE 3.A.18; TIP-OVER EVENT OVERPACK SLAMS AGAINST THE FOUNDATION DEVELOPING A RESISTIVE FORCE

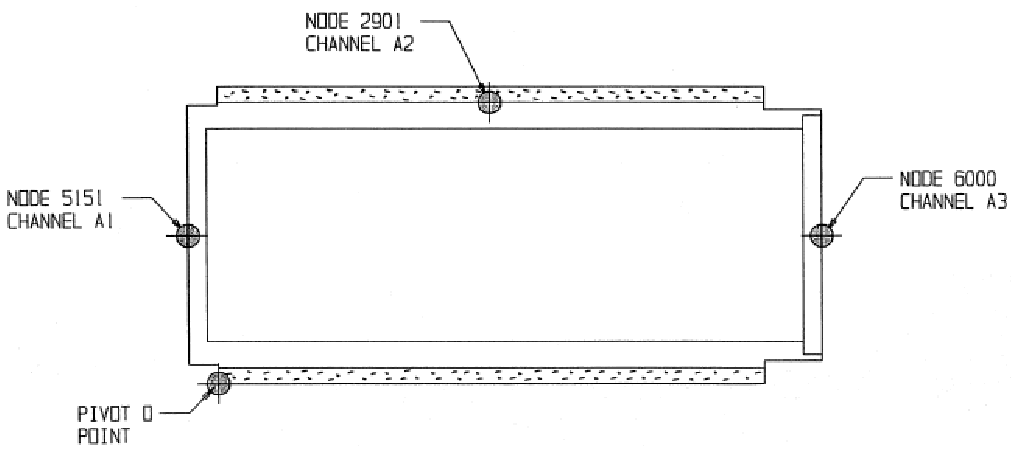

FIGURE 3.A.19; MEASUREMENT POINTS AND CORRESPONDING FINITE-ELEMENT MODEL NODES

CHAPTER 4 THERMAL EVALUATION†

4.0 INTRODUCTION

The HI-STAR 100 System is designed for the long-term storage of spent nuclear fuel (SNF) in either a vertical or horizontal position. An array of HI-STAR 100 Systems regularly spaced on a square or rectangular pitch will be stored on a concrete ISFSI pad in an open environment. In this section, compliance of the HI-STAR 100 thermal performance to 10CFR72 requirements for storage under normal conditions is established. The analysis considers passive rejection of decay heat from the stored SNF assemblies to the environment under the most severe design basis ambient conditions. Effects due to incident solar radiation as well as partial radiation blockage due to the presence of neighboring casks at an ISFSI site are included in the analyses.

The guidelines presented in NUREG-1536 [4.1.3] include specific acceptance criteria that should be fulfilled by the cask thermal design. The applicable criteria are summarized here as follows:

1. The fuel cladding temperature should be below the applicable temperature limit (see Section 4.3).

2. The fuel cladding temperature should be maintained below 570ºC (1058ºF) for short-term accident and short-term off-normal conditions. For fuel transfer conditions the fuel cladding temperature should be maintained below 570ºC (1058ºF) for MBF and below 400ºC (752ºF) for HBF.

3. The maximum internal pressure of the cask should remain within its design pressures for normal (1% rod rupture), off-normal (10% rod rupture), and accident (100% rod rupture) conditions.

4. The cask and fuel materials should be maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions.

5. Deleted.

6. Deleted.

7. The cask system should be passively cooled.

8. The thermal performance of the cask should be within the allowable design criteria specified in FSAR Chapters 2 and 3 for normal, off-normal, and accident conditions.

As demonstrated in this chapter, the HI-STAR 100 System is designed to comply with all criteria listed above. All thermal analyses to evaluate the normal condition performance of a HI-STAR

† Wherever multiple units are shown, the US units are the governing value, and the SI units are shown for information only.

100 System are described in Section 4.4. All analyses for off-normal conditions are described in Section 11.1. All analyses for accident conditions are described in Section 11.2. Section 4.2 lists the material properties data required to perform the thermal analyses and Section 4.3 provides the applicable temperature limits criteria required to demonstrate the adequacy of the HI-STAR 100 System design under all conditions.

The safety analyses summarized in this chapter demonstrate acceptable margins to the allowable limits under all design basis loading conditions and operational modes. Minor changes to the design parameters that inevitably occur during the product's life cycle which are treated within the purview of 10CFR72.48 and are ascertained to have an insignificant effect on the computed safety factors may not prompt a formal reanalysis and revision of the results and associated data in the tables of this chapter unless the cumulative effect of all such un-quantified changes on the reduction of any of the computed safety margins cannot be deemed to be insignificant. For purposes of this determination, an insignificant loss of safety margin with reference to an acceptance criterion is defined as the estimated reduction that is no more than one order of magnitude below the available margin reported in the FSAR. To ensure rigorous configuration control, the information in the Licensing drawings in Section 1.5 should be treated as the authoritative source for numerical analysis at all times. Reliance on the input data and associated results in this chapter for additional mathematical computations may not be appropriate as they serve the sole purpose of establishing safety compliance in accordance with the acceptance criteria set down in Chapter 2 and in this chapter.

Some of the new information in this chapter is directly extracted from previously NRC approved Holtec dockets; this information is shown in *italics.* In Chapter 4, this information was extracted from References [4.0.1] and [4.0.2]. All changes in this revision are marked with revision bars.

TABLE 4.0.1

HI-STAR 100 TRANSPORT SAR MATERIAL GERMANE TO
THE EVALUATIONS IN THIS FSAR

Location in This Storage FSAR	Subject of Reference	Location in HI-STAR 100 Transport SAR
Subparagraph 4.4.1.1.4	Fuel Basket In-Plane Conductive Heat Transport	Subparagraph 3.4.1.1.4
Subparagraph 4.4.1.1.5	Heat Transfer in Fuel Basket Peripheral Regions	Subparagraph 3.4.1.1.5
Subparagraph 4.4.1.1.7	Heat Transfer from Overpack Peripheral Surfaces	Subparagraph 3.4.1.1.7
Subparagraph 4.4.1.1.11	Fluent Model for Temperature Field Computation	Subparagraph 3.4.1.1.12
Paragraph 4.4.2.1	Maximum Temperatures Under Normal Storage Conditions	Figures 3.4.16 & 3.4.17 Tables 3.4.10 & 3.4.11
Subsection 4.4.4	Maximum Internal Pressure	Tables 3.4.13 & 3.4.15
Subsection 4.4.5	Maximum Thermal Stress	Table 3.4.24

4.1 DISCUSSION

A sectional view of the HI-STAR 100 dry storage system has been presented earlier (see Figure 1.2.1). The system consists of an MPC loaded into an overpack with a bolted closure plate. The fuel assemblies reside inside the MPC which is sealed with a welded lid to form the confinement boundary. The MPC contains a stainless steel honeycomb basket structure which provides square-shaped fuel compartments (called boxes) of appropriate dimensions to facilitate insertion of fuel assemblies prior to welding of the lid. Each box panel (except the periphery panels of the MPC-68) is provided with thermal neutron absorber sandwiched between a sheathing plate and the box panel along the entire length of the active fuel region. Prior to sealing the lid, the MPC is backfilled with helium up to the design basis initial loading (Table 1.2.2). This provides a stable and inert environment for long-term storage of the SNF. Additionally, the annular gap formed between the MPC and the overpack is backfilled with helium of the same quality before the overpack vent and drain port plugs are installed. Heat is transferred from the SNF in a HI-STAR 100 System to the environment by passive heat transport mechanisms only.

The helium backfill gas is an integral part of the MPC and overpack thermal designs. The helium fills all the spaces between solid components and provides an improved conduction medium (compared to air) for dissipating decay heat in the MPC. Additionally, helium in the spaces between the fuel basket and the MPC shell is heated differentially and, therefore, subject to the "Rayleigh" effect which is discussed in detail later. To ensure that the helium gas is retained and is not diluted by lower conductivity air, the MPC confinement boundary is designed to comply with the provisions of the ASME B&PV Code Section III, Subsection NB, as an all-seal-welded pressure vessel with redundant closures. Similarly, the overpack helium retention boundary is designed as an ASME B&PV Code Section III, Subsection NB pressure vessel. Both the MPC confinement boundary and the overpack helium retention boundary are required to meet maximum leakage rate Technical Specifications included in Chapter 12 of this FSAR. These leakage rate criteria are selected to ensure the presence of helium during the entire storage life. It is additionally demonstrated in Subsection 11.1.3 that the failure of one confinement boundary seal, a severe off-normal event, will not result in a breach of the confinement boundary. The helium gas is therefore retained and undiluted, and may be credited in the thermal analyses.

An important thermal design criterion imposed on the HI-STAR 100 System is to limit the maximum fuel cladding temperature to within design basis limits (Table 2.2.3) for long-term storage of design basis fuel assemblies. An equally important design criterion is to reduce temperature gradients within the MPC to minimize thermal stresses. In order to meet these design objectives, the HI-STAR 100 MPC basket is designed to possess certain distinctive characteristics, which are summarized in the following.

The MPC design minimizes resistance to heat transfer within the basket and basket periphery regions. This is ensured by an uninterrupted panel-to-panel connectivity realized in the all-welded honeycomb basket structure. Furthermore, the MPC design incorporates top and bottom plena with interconnected downcomer paths. The top plenum is formed by the gap between the bottom of the MPC lid and the top of the honeycomb fuel basket, and by elongated semicircular holes in each basket cell wall. The bottom plenum is formed by large elongated semicircular holes at the base of all cell walls. The MPC basket is designed to eliminate structural discontinuities (i.e., gaps) which introduce large thermal resistances to heat flow. Consequently, temperature gradients are minimized in the design, which results in lower thermal stresses within the basket. Low thermal stresses are also ensured by an MPC design which permits unrestrained axial and radial growth of the basket to eliminate the possibility of thermally induced stresses due to restraint of free-end expansion.

Finally, it is heuristically apparent from the geometry of the MPC that the basket metal, the fuel assemblies, and the contained helium mass will be at their peak temperatures at or near the longitudinal axis of the MPC. The temperatures will attenuate with increasing radial distance from this axis, reaching their lowest values at the outer surface of the MPC shell. Conduction along the metal walls and radiant heat exchange from the fuel assemblies to the MPC metal mass would therefore result in substantial differences in the bulk temperatures of helium columns in different fuel storage cells. Since two fluid columns at different temperatures in communicative contact cannot remain in static equilibrium, the non-isotropic temperature field in the MPC internal space due to conduction and radiation heat transfer mechanisms guarantees the incipience of the third mode of heat transfer: natural convection.

In vertically-oriented systems, the helium columns traverse the vertical storage cavity spaces, redistributing heat within the MPC. Elongated holes in the bottom of the cell walls, liberal flow space and elongated holes at the top, and wide open downcomers along the outer periphery of the basket ensure a smooth helium flow regime. The most conspicuous beneficial effect of the helium thermosiphon circulation, as discussed above, is the mitigation of internal thermal stresses in the MPC. Another beneficial effect is reduction of the peak fuel cladding temperatures of the fuel assemblies located in the interior of the basket. However, in the interest of conservatism, no credit for the thermosiphon action is taken in the thermal analysis reported in this chapter. To partially compensate for the reduction in the computed heat rejection capability due to the complete neglect of the global thermosiphon action within the MPC, flexible heat conduction elements made of aluminum are interposed in the large peripheral spaces between the MPC shell and the fuel basket in some instances. These heat conduction elements, shown in the MPC Drawings in Section 1.5, are engineered to possess lateral flexibility such that they can be installed in the peripheral spaces to create a nonstructural thermal connection between the basket and the MPC shell. In their installed condition, the heat conduction elements will conform to and contact the MPC shell and the basket walls. MPC manufacturing procedures have been established to ensure that the thermal design objectives for the conduction elements set forth in this document are realized in the actual hardware.

Three distinct MPC basket geometries are included in the HI-STAR 100 System for storage of PWR and BWR SNF assemblies. For intact PWR fuel storage, a 24-assembly design is depicted in Figure 1.2.4 and a 32-assembly design is depicted in Figure 1.2.3. A 68-assembly design for storage of intact or damaged BWR fuel is shown in Figure 1.2.2. Damaged BWR fuel and fuel debris must comply with design basis characteristics listed in Table 2.1.7 to allow storage in the MPC-68 and MPC-68F, respectively. Each basket design must comply with the applicable temperature limits for normal, off-normal and accident conditions under the imposed heat generation loads from stored fuel assemblies.

The design basis intact PWR and BWR decay heat per assembly and the MPC total decay heat load for the three basket configurations (i.e., MPC-24, MPC-32, and MPC-68) are stated in Tables 2.1.6 for PWR fuel storage and 1.2.2 for BWR fuel storage. Table 2.1.7 lists the design basis thermal requirements for damaged fuel assemblies. Table 2.1.11 lists the design basis thermal requirements for stainless steel clad fuel assemblies for storage in the MPCs. The HI-STAR 100 System consisting of the overpack and MPCs under normal storage conditions at an ISFSI pad is conservatively analyzed for the limiting design basis heat loads.

Thermal analysis of the HI-STAR 100 System is based on including all three fundamental modes of heat transfer: conduction, natural convection and radiation. Different combinations of these modes are active in different regions and different orientations of the system. These modes are properly identified and conservatively analyzed within each region of the MPC and overpack to enable bounding calculations of the temperature distribution within the HI-STAR 100 System.

On the outside surface of the overpack, heat is dissipated to the environment by buoyancy induced convective air flow (natural convection) and thermal radiation. In the overpack internal metal structure, only conductive heat transport is possible. Between metal surfaces (e.g., between neighboring fuel rod surfaces) heat transport is due to a combination of conduction through a gaseous medium (helium) and thermal radiation. The heat transfer between the fuel basket external surface and the MPC shell's inner surface is further influenced by the so-called "Rayleigh" effect. However, in the interest of conservatism, the most potent heat transport mechanism, the buoyancy induced thermosiphon which occurs within the MPC basket (aided by the MPC design which provides low pressure drop helium flow recirculation loops formed by the fuel cells, top plenum, downcomers and bottom plenum) is neglected.

The total heat generation in each assembly is non-uniformly distributed over the active fuel length to account for the design basis fuel burnup distribution listed in Chapter 2 (Table 2.1.8). As discussed later in this chapter (Subsection 4.4.6), an array of conservative assumptions bias the results of the thermal analysis towards much reduced computed margins than would be obtained by a rigorous analysis of the problem.

4.2 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

Materials used in the HI-STAR 100 System include stainless steels (Alloy X), carbon steels, Holtite-A neutron shield, neutron absorbers, aluminum alloy 1100 heat conduction elements (in some instances*), and helium. In Table 4.2.1, a summary of references used to obtain cask material properties for performing all thermal analyses is presented.

Thermal conductivities of the constituent Alloy X steels and the bounding Alloy X thermal conductivity are reported in Appendix 1.A of this report. Tables 4.2.2, 4.2.3 and 4.2.9 provide numerical thermal conductivity data of materials at several representative temperatures. The neutron absorber materials Boral and Metamic are both made of aluminum powder and boron carbide powder. Although their manufacturing processes differ, from a thermal standpoint, their ability to conduct heat is virtually identical. Therefore, the values of conductivity of the original neutron absorber (Boral) continue to be used in the thermal calculations. Table 4.2.8 lists the thermal properties of Boral components (i.e., B_4C core and aluminum cladding materials). Surface emissivity data for key materials of construction is provided in Table 4.2.4.

The emissivity properties of painted external cask surfaces are generally excellent. Kern [4.2.5] reports an emissivity range of 0.8 to 0.98 for a wide variety of paints. In the HI-STAR 100 thermal analysis, an emissivity of 0.85† is applied to external painted surfaces. A conservative solar absorptivity coefficient of 1.0 is applied to all exposed cask surfaces.

In Table 4.2.5, the heat capacity and density of different cask materials are presented. These properties are used in performing transient (i.e., hypothetical fire accident condition) analyses. Table 4.2.6 provides viscosity data on the helium gas.

The overpack outside surface heat transfer coefficient is calculated by accounting for both natural convection heat transfer and radiation. The natural convection coefficient depends upon the product of Grashof (Gr) and Prandtl (Pr) numbers. Following the approach developed by Jakob and Hawkins [4.2.9], the product Gr×Pr is expressed as $L^3\Delta TZ$, where L is the height of the cask, ΔT is the overpack surface-to-ambient temperature differential and Z is a parameter which depends upon air properties (which are known functions of temperature) evaluated at the average film temperature. The temperature dependence of Z for air is provided in Table 4.2.7.

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* In MPC-32, aluminum heat conduction inserts are optional equipment available by user request. The HI-STAR 100 transportation SAR, being incorporated by reference for the storage of horizontally-oriented systems with MPC-32, permits (but does not require) aluminum heat conduction inserts in the basket-to-shell peripheral spaces. The aluminum heat conduction inserts provide a high-conductivity heat transfer path from the outer surfaces of the fuel basket to the inner surface of the enclosure vessel shell. Their presence would improve heat transfer from the fuel basket to the enclosure vessel shell.

† This is conservative with respect to prior cask industry practice, which has historically accepted higher emissivities. For example, a higher emissivity for painted surfaces (ε = 0.95) is used in the TN-32 cask TSAR (Docket 72-1021).

Table 4.2.1

SUMMARY OF HI-STAR 100 SYSTEM MATERIALS
THERMAL PROPERTY REFERENCES

Material	Emissivity	Conductivity	Density	Heat Capacity
Helium	N/A	Handbook [4.2.2]	Ideal Gas Law	Handbook [4.2.2]
Air	N/A	Handbook [4.2.2]	Ideal Gas Law	Handbook [4.2.2]
Zircaloy	EPRI [4.2.3]	NUREG [4.2.6], [4.2.7]	Rust [4.2.4]	Rust [4.2.4]
UO_2	Not Used	NUREG [4.2.6], [4.2.7]	Rust [4.2.4]	Rust [4.2.4]
Stainless steel	Kern [4.2.5]	ASME [4.2.8]	Marks [4.2.1]	Marks [4.2.1]
Carbon steel	Kern [4.2.5]	ASME [4.2.8]	Marks [4.2.1]	Marks [4.2.1]
Aluminum Alloy 1100 (Heat Conduction Elements)	Handbook [4.2.2]	ASME [4.2.8]	ASME [4.2.8]	ASME [4.2.8]
Boral†	Not Used	Test Data	Test Data	Test Data
Metamic	Not Used	Test Data [4.2.10], [4.2.11]	Test Data [4.2.10], [4.2.11]	Test Data [4.2.10], [4.2.11]
Holtite-A††	Not Used	Conservative Bounding Values	See Footnote	See Footnote

† AAR Structures Boral thermophysical test data.

†† The Holtite-A thermophysical properties (density, ρ, and heat capacity, c_p) were selected to conservatively understate the neutron shield thermal inertia (product of ρ and c_p) in the fire accident evaluation (see Table 4.2.5).

Table 4.2.2

SUMMARY OF HI-STAR 100 SYSTEM MATERIALS THERMAL CONDUCTIVITY DATA

Material	Btu/ft-hr-°F [W/m-K]		
	@ 200°F (93.3°C)	**@ 450°F (232.2°C)**	**@ 700°F (371.1°C)**
Helium	0.0976 [0.169]	0.1289 [0.223]	0.1575 [0.273]
Air	0.0173 [0.030]	0.0225 [0.039]	0.0272 [0.047]
Alloy X	8.4 [14.5]	9.8 [17.0]	11.0 [19.0]
Carbon Steel Radial Connectors	29.2 [50.5]	27.1 [46.9]	24.6 [42.6]
Carbon Steel Gamma Shield Layers	24.4 [42.2]	23.9 [41.4]	22.4 [38.8]
Holtite-A†	See Footnote	See Footnote	See Footnote
Cryogenic Steel	23.8 [41.2]	23.7 [41.0]	22.3 [38.6]

† No credit taken for conduction through Holtite-A for the steady-state analysis. Before and after fire conditions for fire accident analysis (i.e., the conductivity is conservatively set equal to zero). A conductivity of 1.0 Btu/ft-hr-°F (1.73 W/m-K) is conservatively applied during fire condition.

Table 4.2.3

SUMMARY OF FUEL ELEMENT COMPONENTS THERMAL CONDUCTIVITY DATA

Zircaloy Cladding		Fuel (UO_2)	
Temperature (°F [°C])	**Conductivity (Btu/ft-hr-°F [W/m-K])**	**Temperature (°F [°C])**	**Conductivity (Btu/ft-hr-°F [W/m-K])**
392 [200]	8.28† [14.3]	100 [38]	3.48 [6.0]
572 [300]	8.76 [15.2]	448 [256]	3.48 [6.0]
752 [400]	9.60 [16.6]	570 [299]	3.24 [5.6]
932 [500]	10.44 [18.1]	793 [423]	2.28† [3.9]

† Lowest value of conductivity is used in the thermal analysis for conservatism.

Table 4.2.4

SUMMARY OF MATERIALS SURFACE EMISSIVITY DATA

Material	Emissivity
Zircaloy cladding	0.80
Painted surfaces	0.85
Rolled carbon steel	0.66
Stainless steel	0.36
Sandblasted aluminum	0.40

Table 4.2.5

MATERIALS DENSITY AND HEAT CAPACITY PROPERTIES SUMMARY

Material	Density (lbm/ft^3 [kg/m^3])	Heat Capacity (Btu/lbm-°F [J/kg-K])
Helium	(Ideal Gas Law)	1.24 [5192]
Zircaloy cladding	409 [6552]	0.0728 [305]
Fuel (UO_2)	684 [10957]	0.056 [234]
Carbon steel	489 [7833]	0.1 [419]
Stainless steel	501 [8025]	0.12 [502]
Boral Neutron Absorber	154.7 [2478]	0.13 [544]
Metamic Neutron Absorber	163.4 [2617]	0.22 [921]
Aluminum Alloy 1100	169.9 [2722]	0.23 [963]
Holtite-A	105.0 [1682]	0.39 [1633]

Table 4.2.6

HELIUM GAS VISCOSITY† VARIATION WITH TEMPERATURE

Temperature (ºF [ºC])	Viscosity (Micropoise [kg/m-s])
167.4 [75.2]	220.5 [2.2×10^{-5}]
200.3 [93.5]	228.2 [2.3×10^{-5}]
297.4 [147.4]	250.6 [2.5×10^{-5}]
346.9 [174.9]	261.8 [2.6×10^{-5}]
463.0 [239.4]	288.7 [2.9×10^{-5}]
537.8 [281.0]	299.8 [3.0×10^{-5}]
737.6 [392.0]	338.8 [3.4×10^{-5}]

† Obtained from Rohsenow and Hartnett [4.2.2].

Table 4.2.7

VARIATION OF NATURAL CONVECTION PROPERTIES
PARAMETER "Z" FOR AIR WITH TEMPERATURE†

Temperature (^oF [^oC])	**Z ($ft^{-3}\,{}^oF^{-1}$ [$m^{-3}\,K^{-1}$])**
40 [4]	2.1×10^6 [133.5×10^6]
140 [60]	9.0×10^5 [57.2×10^6]
240 [116]	4.6×10^5 [29.2×10^6]
340 [171]	2.6×10^5 [16.5×10^6]
440 [227]	1.5×10^5 [9.5×10^6]

† Obtained from Jakob and Hawkins [4.2.9]

Table 4.2.8

BORAL COMPONENT MATERIALS[†]
THERMAL CONDUCTIVITY DATA

Temperature (°F [°C])	B_4C Core Conductivity (Btu/ft-hr-°F [W/m-K])	Aluminum Cladding Conductivity (Btu/ft-hr-°F [W/m-K])
212 [100]	48.09 [83.2]	100.00 [173.1]
392 [200]	48.03 [83.1]	104.51 [180.9]
572 [300]	47.28 [81.8]	108.04 [187.0]
752 [400]	46.35 [80.2]	109.43 [189.4]

† Both B_4C and aluminum cladding thermal conductivity values are obtained from AAR Structures Boral thermophysical test data.

Table 4.2.9

HEAT CONDUCTION ELEMENTS (ALUMINUM ALLOY 1100)
THERMAL CONDUCTIVITY DATA

Temperature (°F [°C])	**Conductivity (Btu/ft-hr-°F [W/m-K])**
100 [38]	131.8 [228.1]
200 [93]	128.5 [222.4]
300 [149]	126.2 [218.4]
400 [204]	124.5 [215.5]

4.3 SPECIFICATIONS FOR COMPONENTS

HI-STAR 100 System materials and components designated as "Important to Safety" (i.e., required to be maintained within their safe operating temperature ranges to ensure their intended function) which warrant special attention are summarized in Table 4.3.1. Long-term stability and continued neutron shielding ability of Holtite-A neutron shield material under normal storage conditions are ensured when material exposure temperatures are maintained below the maximum allowable limit. The integrity of the overpack helium retention boundary is assured by maintaining the temperature of the mechanical seals within the manufacturer's recommended operating temperature limits. Long-term integrity of SNF is ensured by the HI-STAR 100 System thermal performance, which demonstrates that fuel cladding temperatures are maintained below design basis limits. Neutron absorber materials used in MPC baskets for criticality control (composite materials composed of B_4C and aluminum) are stable up to 1000°F (538°C) for short-term and 850°F (454°C) for long-term dry storage†. However, for conservatism, a significantly lower maximum temperature limit is imposed.

Compliance to 10CFR72 requires, in part, identification and evaluation of short-term off-normal and severe hypothetical accident conditions. The inherent mechanical stability characteristics of cask materials and components ensure that no significant functional degradation is possible due to exposure to short-term temperature excursions outside the normal long-term temperature limits. For evaluation of HI-STAR 100 System thermal performance under off-normal or hypothetical accident conditions, material temperature limits for short-duration events are provided in Table 4.3.1. Fuel temperature limits mandated by ISG-11 [4.3.8] are adopted for evaluation of cladding integrity under normal, short term operations, off-normal and accident conditions. These limits are applicable to all fuel types, burnup levels and cladding materials approved by the NRC for power generation.

It is recognized that hydrides present in irradiated fuel rods (predominantly circumferentially oriented) dissolve at cladding temperatures above 400°C [4.3.9]. Upon cooling below a threshold temperature (T_P), the hydrides precipitate and reorient to an undesirable (radial) direction if cladding stresses at the hydride precipitation temperature T_P are excessive. For moderate burnup fuel, T_P is conservatively estimated as 350°C [4.3.9]. In a recent study, PNNL has evaluated a number of bounding fuel rods for reorientation under hydride precipitation temperatures for MBF [4.3.9]. The study concludes that hydride reorientation is not credible during short-term operations involving low to moderate burnup fuel (up to 45 GWD/MTU). Accordingly, the higher ISG-11 temperature limit is justified for moderate burnup fuel and is adopted for short-term operations for MBF fueled MPCs (see Table 4.3.1).

† AAR Structures Boral thermophysical test data.

Table 4.3.1

HI-STAR 100 SYSTEM MATERIAL TEMPERATURE LIMITS

Material	**Normal Long-Term Temperature Limits (°F [°C])**	**Short-Term Temperature Limits (°F [°C])**
Fuel cladding (Zircaloy and stainless steel)	752 [400]	1058 [570] for MBF 752 [400] for HBF
Neutron Absorber	800 [427]	950 [510]
Overpack closure plate mechanical seal, vent and drain port plug seals	See Table 2.2.3	See Table 2.2.3
Holtite-A[††]	300 [149]	300 [149]

[††] See Appendix 1.B.

Table 4.3.2

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Table 4.3.3

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Table 4.3.4

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Table 4.3.5

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Table 4.3.6

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4.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF STORAGE

4.4.1 Thermal Model

The HI-STAR 100 MPC basket designs consist of three distinct geometries to hold 24 PWR, 32 PWR or 68 BWR fuel assemblies. The basket is a matrix of square compartments (called boxes) to hold the fuel assemblies. The basket is a honeycomb structure of Alloy X plates with full-length edge-welded intersections to form an integral basket configuration. Individual cell walls (except outer periphery MPC-68 cell walls) are provided with neutron absorber panels sandwiched between the box wall and a sheathing plate over the full length of the active fuel region.

The design basis decay heat generation (per PWR or BWR assembly) for long-term normal storage is specified in Table 2.1.6. The decay heat is conservatively considered to be non-uniformly distributed over the active fuel length based on the design basis axial burnup distribution provided in Chapter 2 (Table 2.1.8).

Transport of heat from the interior of the MPC basket to its outer periphery is accomplished by a combination of conduction through the MPC basket metal grid structure, conduction and radiation heat transfer in the relatively small helium gaps between the fuel assemblies and basket cell walls, and radiation and conduction from the fuel basket periphery to the MPC shell. Heat dissipation across the gap between the MPC basket periphery and the MPC shell is by a combination of helium conduction, natural convection (by means of the “Rayleigh” effect), radiation across the gap, and conduction in the aluminum alloy 1100 heat conduction elements (in some instances). Between the MPC exterior and the overpack interior is a small clearance region which is evacuated and backfilled with helium. Helium, besides being inert, is a better heat conduction medium than air. Thus, heat conduction through the MPC/overpack helium gap will minimize temperature differentials across this region.

The overpack, under normal storage conditions, passively rejects heat to the outside environment. Cooling of the outside overpack surfaces is by natural convection and thermal radiation and, for vertically-oriented casks, the bottom surface conducts heat through the ISFSI concrete pad to the ground. Analytical modeling details of the various thermal transport mechanisms are provided in the following.

4.4.1.1 Analytical Model - General Remarks

Transport of heat from the heat generation region (fuel assemblies) to the outside environment (ambient air or ground) is analyzed broadly in terms of three interdependent thermal models. The first model considers transport of heat from the fuel assembly to the basket cell walls. This model recognizes the combined effects of conduction (through helium) and radiation, and is essentially a finite element technology based update of the classical Wooton & Epstein [4.4.1] (which considered radiative heat exchange between fuel rod surfaces) formulation. The second model considers heat transport within an MPC cross section by conduction and radiation. The effective cross sectional thermal conductivity of the basket and basket periphery regions, obtained from a combined fuel assembly/basket heat conduction-radiation model developed on

ANSYS [4.1.1], are applied to an axisymmetric thermal model of the HI-STAR 100 System on the FLUENT [4.1.2] code. The third model deals with the transmission of heat from the MPC exterior surface to the external environment (heat sink). From the MPC shell to the overpack exterior surface, heat is conducted through an array of layered shells representing the MPC-to-overpack helium gap, overpack inner shell, intermediate shells, Holtite-A and overpack outer shell. Heat rejection from the outside cask surfaces to ambient air is considered by accounting for natural convection and thermal radiation heat transfer mechanisms from the exposed surfaces. For vertically-oriented casks the bottom overpack face, in contact with the ISFSI pad, rejects a small quantity of heat by conduction through the pad to the ground. The reduction in radiative heat exchange between cask outside surfaces and ambient air because of blockage from the neighboring casks arranged for normal storage in a regular array on the ISFSI pad is recognized in the analysis. The exposed overpack outer surfaces are modeled as heated surfaces in convective and radiative heat exchange with air and as a recipient of heat input through insolation. Insolation on the cask surfaces is based on 12-hour levels prescribed in 10CFR71, averaged over a 24-hour period, after accounting for partial blockage conditions.

Subparagraphs 4.4.1.1.1 through 4.4.1.1.11 contain a systematic description of the mathematical models devised to articulate the temperature field in the HI-STAR 100 System. Table 4.4.2 shows the relationship between the mathematical models and the corresponding regions (i.e., fuel, MPC, overpack, etc.) of the HI-STAR 100 System. The description begins with the method to characterize the heat transfer behavior of the prismatic (square) opening referred to as the "fuel space" with a heat emitting fuel assembly situated in it. The methodology utilizes a finite element procedure to replace the heterogeneous SNF/fuel space region with an equivalent solid body having a well-defined temperature-dependent conductivity. In the following subparagraph, the method to replace the "composite" walls of the fuel basket cells with an equivalent "solid" wall is presented. Having created the mathematical equivalents for the SNF/fuel spaces and the fuel basket walls, the method to represent the MPC cylinder containing the fuel basket by an equivalent cylinder whose thermal conductivity is a function of the spatial location and coincident temperature is presented.

Following the approach of presenting descriptions starting from the inside and moving to the outer region of a cask, the next subsections present the mathematical model to simulate the overpack. Subparagraph 4.4.1.1.11 concludes the presentation with a description of how the different models for the specific regions within the HI-STAR 100 System are assembled into the final FLUENT model. Finally, a subparagraph to describe the solution for the special case of vacuum in the MPC space (no helium) is presented.

4.4.1.1.1 Overview of the Thermal Model

Thermal analysis of the HI-STAR 100 System is performed by assuming that the system is subject to its maximum heat duty with each storage location occupied and with the heat generation rate in each stored fuel assembly equal to design basis maximum value. While the assumption of equal heat generation imputes a certain symmetry to the cask thermal problem, the thermal model must incorporate three attributes of the physical problem to perform a rigorous analysis of a fully loaded cask:

i. While the rate of heat conduction through metals is a relatively weak function of temperature, radiation heat exchange is a nonlinear function of surface temperatures.

ii. Heat generation in the MPC is axially non-uniform due to non-uniform axial burnup profile in the fuel assemblies.

iii. Inasmuch as the transfer of heat occurs from inside the basket region to the outside, the temperature field in the MPC is spatially distributed with the maximum values reached in the central core region.

It is clearly impractical to model every fuel rod in every stored fuel assembly explicitly. Instead, the cross section bounded by the inside of the storage cell, which surrounds the assemblage of fuel rods and the interstitial helium gas, is replaced with an "equivalent" square (solid) section characterized by an effective thermal conductivity. Figure 4.4.1 pictorially illustrates the homogenization concept. Further details of this procedure for determining the effective conductivity are presented in Subparagraph 4.4.1.1.2; it suffices to state here that the effective conductivity of the cell space will be a function of temperature because the radiation heat transfer (a major component of the heat transport between the fuel rods and the surrounding basket cell metal) is a strong function of the temperatures of the participating bodies. Therefore, in effect, every storage cell location will have a different value of effective conductivity (depending on the coincident temperature) in the homogenized model. The temperature-dependent fuel assembly region effective conductivity is determined by a finite volume procedure, as described in Subparagraph 4.4.1.1.2.

In the next step of homogenization, a planar section of MPC is considered. With each storage cell inside space replaced with an equivalent solid square, the MPC cross section consists of a metallic gridwork (basket cell walls with each square cell space containing a solid fuel cell square of effective thermal conductivity, which is a function of temperature) circumscribed by a circular ring (MPC shell). There are multiple distinct materials in this section, namely the homogenized fuel cell squares, the Alloy X structural materials in the MPC (including sheathing), neutron absorber, alloy 1100 aluminum heat conduction elements (in some instances), and helium gas. Each of the constituent materials in this section has a different conductivity. It is emphasized that the conductivity of the homogenized fuel cells is a strong function of temperature.

In order to replace this thermally heterogeneous MPC section with an equivalent conduction-only region, resort to the finite element procedure is necessary. Because the rate of transport of heat within the MPC is influenced by radiation, which is a temperature-dependent effect, the equivalent conductivity of the MPC region must also be computed as a function of temperature. Finally, it is recognized that the MPC section consists of two discrete regions, namely, the basket region and the peripheral region. The peripheral region is the space between the peripheral storage cells and the MPC shell. This space is essentially full of helium surrounded by Alloy X plates and, in some instances, alloy 1100 aluminum heat conduction elements. Accordingly, as illustrated in Figure 4.4.2 for MPC-68, the MPC cross section is replaced with two homogenized regions with temperature-dependent conductivities. In particular, the effective conductivity of the fuel cells is subsumed into the equivalent conductivity of the basket cross section. The finite element procedure used to accomplish this is described in Subparagraph 4.4.1.1.4. The ANSYS finite element code is the vehicle for all modeling efforts described in the foregoing.

In summary, appropriate finite element models are used to replace the MPC cross section with an equivalent two region homogeneous conduction lamina whose local conductivity is a known function of coincident absolute temperature. Thus, the MPC cylinder containing discrete fuel assemblies, helium, neutron absorber and Alloy X, is replaced with a right circular cylinder whose material conductivity will vary with the radial and axial position as a function of the coincident temperature.

The MPC-to-overpack gap is simply an annular space (with a non-uniform width when the cask is in a horizontal orientation) which is readily modeled with an equivalent conductivity which reflects conduction and radiation modes of heat transfer. The overpack is a radially symmetric structure except for the neutron absorber region which is built from radial connectors and Holtite-A (see Figure 4.4.7). Using the classical equivalence procedure described in Subparagraph 4.4.1.1.9, this region is replaced with an equivalent radially symmetric annular cylinder.

The thermal analysis procedure described above makes frequent use of equivalent thermal properties to ease the geometric modeling of the cask components. These equivalent properties are rigorously calculated values based on detailed evaluations of actual cask system geometries. All these calculations are performed conservatively to ensure a bounding representation of the cask system. This process, commonly referred to as submodeling, yields accurate (not approximate) results. Given the detailed nature of the submodeling process, experimental validation of the individual submodels is not necessary.

In this manner, a HI-STAR 100 System overpack containing a loaded MPC, either standing upright on the ISFSI pad or supported horizontally on the ISFSI pad, is replaced with a right circular cylinder with spatially varying temperature-dependent conductivity. Heat is generated within the basket space in this cylinder in the manner of the prescribed axial burnup distribution. In addition, heat is deposited from insolation on the external surface of the overpack. Under steady state conditions the total heat due to internal generation and insolation is dissipated from the outer cask surfaces by natural convection and thermal radiation to the ambient environment. Details of the elements of mathematical modeling are provided in the following.

4.4.1.1.2 Fuel Region Effective Thermal Conductivity Calculation

Thermal properties of a large number of PWR and BWR fuel assembly configurations manufactured by the major fuel suppliers (i.e., Westinghouse, CE, B&W, and GE) have been evaluated for inclusion in the HI-STAR 100 System thermal analysis. It is noted that PWR fuel assemblies are equipped with removable non-fuel hardware, in particular, control rods which are inserted in guide tube locations for in-core usage. In dry cask storage, PWR fuel is optionally stored with the control rods. The control rods, when inserted in the fuel assemblies, displace gas in the guide tubes replacing it with solid materials (neutron absorbers and metals) which conduct heat much more readily. As a result, dissipation of heat in the fuel assemblies is enhanced by the presence of these control rods. For conservatism, credit for presence of control rods in fuel assemblies is neglected. Bounding PWR and BWR fuel assembly configurations are determined using the simplified procedure described below. This is followed by the determination of temperature-dependent properties of the bounding PWR and BWR fuel assembly configurations to be used for cask thermal analysis using a finite volume (FLUENT) approach.

To determine which of the numerous PWR assembly types listed in Table 4.4.5 should be used in the thermal model for the MPC-24 or MPC-32 fuel basket, we must establish which assembly has the maximum thermal resistance. The same determination must be made for the MPC-68, out of the menu of SNF types listed in Table 4.4.6. For this purpose, we utilize a simplified procedure which is described below.

Each fuel assembly consists of a large array of fuel rods typically arranged on a square layout. Every fuel rod in this array is generating heat due to radioactive decay in the enclosed fuel pellets. There is a finite temperature difference required to transport heat from the innermost fuel rods to the storage cell walls. Heat transport within the fuel assembly is based on principles of conduction heat transfer combined with the highly conservative analytical model proposed by Wooton and Epstein [4.4.1]. The Wooton-Epstein model considers radiative heat exchange between individual fuel rod surfaces as a means to bound the hottest fuel rod cladding temperature.

Transport of heat energy within any cross section of a fuel assembly is due to a combination of radiative energy exchange and conduction through the helium gas which fills the interstices between the fuel rods in the array. With the assumption of uniform heat generation within any given horizontal cross section of a fuel assembly, the combined radiation and conductive heat transport effects result in the following heat flow equation:

$$Q = \sigma \; C_o \; F_\varepsilon \; A \, [T_C^4 - T_B^4] + 13.5740 \; L \; K_{cs} \, [T_C - T_B]$$

where:

$$F_\varepsilon = \text{Emissivity Factor}$$
$$= \frac{1}{\left(\frac{1}{\varepsilon_C} + \frac{1}{\varepsilon_B} - 1\right)}$$

$\varepsilon_C, \varepsilon_B$ = emissivities of fuel cladding, fuel basket (see Table 4.2.4)

$$C_o = \text{Assembly Geometry Factor}$$
$$= \frac{4N}{(N+1)^2} \text{ (when N is odd)}$$
$$= \frac{4}{N+2} \text{ (when N is even)}$$

N = Number of rows or columns of rods arranged in a square array

A = fuel assembly "box" heat transfer area (ft^2)
= $4 \times$ width $\times$ length

L = fuel assembly length (ft)

K_{cs} = fuel assembly constituent materials volume fraction weighted mixture conductivity (Btu/hr-ft-ºF)

T_C = hottest fuel cladding temperature (ºR)

T_B = box temperature (ºR)

Q = net radial heat transport from the assembly interior (Btu/hr)

σ = Stefan-Boltzmann Constant (0.1714×10^{-8} Btu/ft^2-hr-ºR^4)

In the above heat flow equation, the first term is the Wooten-Epstein radiative heat flow contribution while the second term is the conduction heat transport contribution based on the classical solution to the temperature distribution problem inside a square shaped block with uniform heat generation [4.4.5]. The 13.574 factor in the conduction term of the equation is the shape factor for two-dimensional heat transfer in a square section. Planar fuel assembly heat transport by conduction occurs through a series of resistances formed by the interstitial helium fill gas, fuel cladding and enclosed fuel. An effective planar mixture conductivity is determined by a volume fraction weighted sum of the individual constituent material resistances. For BWR assemblies, this formulation is applied to the region inside the fuel channel. A second conduction and radiation model is applied between the channel and the fuel basket gap. These two models are combined, in series, to yield a total effective conductivity.

The effective conductivity of the fuel for several representative PWR and commonly used BWR assemblies is presented in Tables 4.4.5 and 4.4.6. At higher temperatures (approximately 450ºF and above), the zircaloy clad fuel assemblies with the lowest effective thermal conductivities are the W-17×17 OFA (PWR) and the GE11-9×9 (BWR). A discussion of fuel assembly conductivities for some of the newer 10×10 array and plant specific BWR fuel designs is presented near the end of this subsection. As noted in Table 4.4.6, the Dresden 1 (intact and damaged) fuel assemblies are excluded from consideration. The design basis decay heat load for Dresden-1 intact and damaged fuel (Table 2.1.7) is approximately 58% lower than the MPC-68 design basis maximum heat load (Table 2.1.20). Examining Table 4.4.6, the effective conductivity of the damaged Dresden-1 fuel assembly in a damaged fuel container is approximately 40% lower than the bounding (GE-11 9×9) fuel assembly. Consequently, the fuel cladding temperatures in the HI-STAR 100 System with Dresden-1 intact of damaged fuel assemblies will be bounded by design basis fuel cladding temperatures. This is demonstrated in Subparagraph 4.4.1.1.16. Based on this simplified analysis, the W-17×17 OFA PWR and GE11-9×9 BWR fuel assemblies are determined to be the bounding configurations for analysis of zircaloy clad fuel at design basis maximum heat loads. As discussed in Subsection 4.3.1, stainless clad fuel assemblies with significantly lower decay heat emission characteristics are not deemed to be bounding.

Having established the governing (most resistive) PWR and BWR SNF types, a finite volume code is used to determine the effective conductivities in a conservative manner. Detailed conduction-radiation finite volume models of the bounding PWR and BWR fuel assemblies are developed in the FLUENT code as shown in Figures 4.4.8 and 4.4.9, respectively. The PWR model was originally developed on the ANSYS code which enables individual rod-to-rod and rod-to-basket wall view factor calculations to be performed by the AUX12 procedure for the special case of black body radiation (surfaces emissivity = 1). Limitations of radiation modeling techniques implemented in ANSYS do not permit taking advantage of quarter symmetry of the fuel assembly geometry. Unacceptably long CPU time and large workspace requirements necessary for performing gray body radiation calculations for a complete fuel assembly geometry on ANSYS prompted the development of an alternate simplified model on the FLUENT code. The FLUENT model is benchmarked with the ANSYS model results for a Westinghouse 17×17 fuel assembly geometry for the case of black body radiation (emissivities = 1). The FLUENT model is found to yield conservative results in comparison to the ANSYS model for the "black" surface case. The FLUENT model benchmarked in this manner is used to solve the gray body radiation problem to provide the necessary results for determining the effective thermal conductivity of the governing PWR fuel assembly. The same modeling approach using FLUENT is then applied to the governing BWR fuel assembly, and the effective conductivity of GE11-9×9 fuel determined.

The combined fuel rods-helium matrix is replaced by an equivalent homogeneous material which fills the basket opening by the following two-step procedure. In the first step, the FLUENT-based fuel assembly model is solved by applying equal heat generation per unit length to the individual fuel rods and a uniform boundary temperature along the basket cell opening inside periphery. The temperature difference between the peak cladding and boundary temperatures is used to determine an effective conductivity as described in the next step. For this purpose, we consider a two-dimensional cross section of a square shaped block of size equal to 2L and a uniform volumetric heat source (q_g) cooled at the periphery with a uniform boundary temperature. Under the assumption of constant material thermal conductivity (K), the temperature difference (ΔT) from the center of the cross section to the periphery is analytically given by [4.4.5]:

$$\Delta T = 0.29468 \frac{q_g L^2}{K}$$

This analytical formula is applied to determine the effective material conductivity from a known quantity of heat generation applied in the FLUENT model (smeared as a uniform heat source, q_g) basket opening size and ΔT calculated in the first step.

As discussed earlier, the effective fuel space conductivity is a function of the temperature coordinate. The above two step analysis is carried out for a number of reference temperatures. In this manner, the effective conductivity as a function of temperature is established.

In Table 4.4.23, 10×10 array type BWR fuel assembly conductivity results from a simplified analysis are presented to determine the most resistive fuel assembly in this class. From the data in Table 4.4.23, the Atrium-10 fuel type is determined to be the most resistive in this class of fuel assemblies. A detailed finite element model of this assembly type was developed to rigorously quantify the heat dissipation characteristics. The results of this study are presented in Table 4.4.24 and compared to the BWR bounding fuel assembly conductivity depicted in Figure 4.4.14. The results of this study demonstrate that the bounding fuel assembly conductivity is conservative with respect to the 10x10 class of BWR fuel assemblies.

Table 4.4.25 summarizes plant specific fuel types effective conductivities. From these analytical results, the SPC-5 is determined to be the most resistive fuel assembly in this group of fuel types. A rigorous finite element model of SPC-5 fuel assembly was developed to confirm that its in-plane heat dissipation characteristics are bounded from below by the Design Basis BWR fuel conductivities used in the HI-STAR thermal analysis.

Temperature-dependent effective conductivities of PWR and BWR design basis fuel assemblies (most resistive SNF types) are shown in Figure 4.4.14. The finite volume computational results are also compared to results reported from independent technical sources. From this comparison, it is readily apparent that FLUENT-based fuel assembly conductivities are conservative. The FLUENT computed values (not the published literature data) are used in the MPC thermal analysis presented in this document.

4.4.1.1.3 Effective Thermal Conductivity of Absorber/Sheathing/Box Wall Sandwich

Each MPC basket cell wall (except the MPC-68 outer periphery cell walls) is manufactured with a neutron absorber plate for criticality control. Each neutron absorber plate is sandwiched in a sheathing-to-basket wall pocket. A schematic of the "Box Wall-Absorber-Sheathing" sandwich geometry of an MPC basket is illustrated in Figure 4.4.3. During fabrication, a uniformly normal pressure is applied to each "Box Wall-Absorber-Sheathing" sandwich in the assembly fixture during stitch-welding of the sheathing periphery on the box wall. This ensures adequate surface-to-surface contact for elimination of any macroscopic air gaps. The mean coefficient of linear expansion of the absorber is higher than the thermal expansion coefficients of the basket and sheathing materials. Consequently, basket heat-up from the stored SNF will further ensure a tight fit of the absorber plate in the sheathing-to-box pocket. The presence of small microscopic gaps due to less than perfect surface finish characteristics requires consideration of an interfacial contact resistance between the absorber and the box-sheathing surfaces. A conservative contact resistance resulting from a 2 mil absorber to pocket gap is applied in the analysis. In other words, no credit is taken for the interfacial pressure between the absorber and stainless plate/sheet stock produced by the fixturing and welding process.

Heat conduction properties of a composite "Box Wall-Absorber-Sheathing" sandwich in the two principal basket cross sectional directions as illustrated in Figure 4.4.3 (i.e., lateral "out-of-plane" and longitudinal "in-plane") are unequal. In the lateral direction, heat is transported across layers of sheathing, helium gap, absorber (B_4C and cladding layers) and box wall resistances which are essentially in series (except for the small helium filled end regions shown in Figure 4.4.4). Heat conduction in the longitudinal direction, in contrast, is through an array of essentially parallel resistances comprised of these several layers listed above. For the ANSYS based MPC basket thermal model, corresponding non-isotropic effective thermal conductivities in the two orthogonal sandwich directions are determined and applied in the analysis.

These non-isotropic conductivities are determined by constructing two-dimensional finite-element models of the composite “Box Wall-Absorber-Sheathing” sandwich in ANSYS. A fixed temperature is applied to one edge of the model and a fixed heat flux is applied to the other edge, and the model is solved to obtain the average temperature of the fixed-flux edge. The equivalent thermal conductivity is obtained using the computed temperature difference across the sandwich as input to a one-dimensional Fourier equation for conduction heat transfer as follows:

$$K_{eff} = \frac{q \times L}{T_h - T_c}$$

where:

K_{eff} = effective thermal conductivity
q = heat flux applied in the ANSYS model
L = ANSYS model heat transfer path length
T_h = ANSYS calculated average edge temperature
T_c = specified edge temperature

The heat transfer path length is the width or thickness of the sandwich, respectively, depending on the direction of transfer (i.e., in-plane or out-of-plane).

4.4.1.1.4 Finite Element Modeling of Basket In-Plane Conductive Heat Transport

The heat rejection capability of each MPC basket design (i.e., MPC-24, MPC-32 and MPC-68) is evaluated by developing a thermal model of the combined fuel assemblies and composite basket walls geometry on the ANSYS finite element code. The ANSYS model includes a geometric layout of the basket structure in which the basket “Box Wall-Absorber-Sheathing” sandwich is replaced by a "homogeneous wall" with an equivalent thermal conductivity. Since the thermal conductivity of the Alloy X material is a weakly varying function of temperature, the equivalent "homogeneous wall" must have a temperature-dependent effective conductivity. Similarly, as illustrated in Figure 4.4.4, the conductivities in the "in-plane" and "out-of-plane" directions of the equivalent "homogeneous wall" are different. Finally, as discussed earlier, the fuel assemblies and the surrounding basket cell openings are modeled as homogeneous heat generating regions with effective temperature dependent in-plane conductivity. The methodology used to reduce the heterogeneous MPC basket - fuel assemblage to an equivalent homogeneous region with effective thermal properties is discussed in the following.

Consider a cylinder of height, L, and radius, r_o, with a uniform volumetric heat source term, q_g, insulated top and bottom faces, and its cylindrical boundary maintained at a uniform temperature, T_c. The maximum centerline temperature (T_h) to boundary temperature difference is readily obtained from classical one-dimensional conduction relationships (for the case of a conducting region with uniform heat generation and a constant thermal conductivity K_s):

$$(T_h - T_c) = q_g\, r_o^2/(4\, K_s)$$

Noting that the total heat generated in the cylinder (Q_t) is $\pi\, r_o^2\, L\, q_g$, the above temperature rise formula can be reduced to the following simplified form in terms of total heat generation per unit length (Q_t/L):

$$(T_h - T_c) = (Q_t / L)/ (4\, \pi\, K_s)$$

This simple analytical approach is employed to determine an effective basket cross-sectional conductivity by applying an equivalence between the ANSYS finite element model of the basket and the analytical case. The equivalence principle employed in the HI-STAR 100 System thermal analysis is depicted in Figure 4.4.2. The 2-dimensional ANSYS finite element model of each MPC basket is solved by applying a uniform heat generation per unit length in each basket cell region and a constant basket periphery boundary temperature, T_c'. Noting that the basket region with uniformly distributed heat sources and a constant boundary temperature is equivalent to the analytical case of a cylinder with uniform volumetric heat source discussed earlier, an effective MPC basket conductivity (K_{eff}) is readily derived from the analytical formula and ANSYS solution leading to the following relationship:

$$K_{eff} = N\, (Q_f'/L) / (4\, \pi\, [T_h' - T_c'])$$

where:

N = number of fuel assemblies

(Q_f'/L) = per fuel assembly heat generation per unit length applied in ANSYS model

T_h' = peak basket cross-section temperature from ANSYS model

PWR and BWR basket ANSYS models are depicted in Figures 4.4.11 and 4.4.12. Please note that many of the basket supports and all shims have been conservatively neglected in the models. This conservative geometry simplification, coupled with the conservative neglect of thermal expansion which would minimize the gaps, yields conservative gap thermal resistances. Temperature-dependent equivalent thermal conductivities of fuel region and composite basket walls, as determined from analysis procedures described earlier, are applied to the ANSYS model. The planar ANSYS conduction model is solved by applying a constant basket periphery temperature with uniform heat generation in the fuel region. Table 4.4.7 summarizes effective thermal conductivity values used in subsequent cask thermal modeling. It should be noted that the planar conductivities calculated as described above are actually higher than those reported in Table 4.4.7, imparting additional conservatism to the subsequent calculations.

For vertically-oriented casks, the effective calculated basket cross sectional conductivity and the effective axial direction effective conductivity is conservatively assumed to be equal in the comprehensive HI-STAR 100 System thermal model (see Subparagraph 4.4.1.1.11). It is recalled that the equivalent thermal conductivity values presented in Table 4.4.7 are lower bound values because, among other elements of conservatism, the effective conductivity of most resistive SNF types (Tables 4.4.5 and 4.4.6) are used in the MPC finite element simulations. For horizontally-oriented casks, the effective calculated basket cross sectional conductivity and the effective axial direction effective conductivity are not always assumed equal. Rather, for some configurations, the calculated basket cross sectional conductivity is combined with a higher axial conductivity to obtain an effective isotropic conductivity (as described in Subparagraph 3.4.1.1.4 of the HI-STAR 100 transport SAR [4.0.1]).

4.4.1.1.5 Heat Transfer in MPC Basket Peripheral Region

Each of the MPC designs for storing PWR or BWR fuel are provided with relatively large regions, formed between the relatively cooler MPC shell and hot basket peripheral panels, filled with helium gas. Heat transfer in these helium-filled regions corresponds to the classical case of heat transfer in a differentially heated closed cavity. Experimental studies of this arrangement have been performed by many investigators, including Eckert and Carlson (Int. J. Heat Mass Transfer, vol. 2, p. 106, 1961 [4.4.8]) and Elder (J. Fluid Mech., vol. 23, p. 77, 1965 [4.4.9]). The peripheral region between the basket and MPC inner surface in a vertically-oriented cask is simulated as a tall fluid-filled cavity of height H formed between two differentially heated surfaces (ΔT) separated by a small distance L. In a closed cavity, an exchange of hot and cold fluids occurs near the top and bottom ends of the cavity, resulting in a net transport of heat across the gap. The rate of heat transfer across the cavity is characterized by a Rayleigh number, Ra_L, defined as:

$$R\,a_L = \frac{C_p \rho^2 g \beta \Delta T L^3}{\mu K}$$

where:

C_p	=	fluid heat capacity
ρ	=	fluid density
g	=	acceleration due to gravity
β	=	coefficient of thermal expansion (equal to reciprocal of absolute temperature for gases)
ΔT	=	temperature difference between the hot and cold surfaces
L	=	spacing between the hot and cold surfaces
μ	=	fluid viscosity
K	=	fluid conductivity

Hewitt et al. [4.4.6] recommends the following Nusselt number correlation for heat transport in tall cavities:

$$Nu_L = 0.42 \; Ra_L^{1/4} \; Pr^{0.012} \; (\frac{H}{L})^{-0.3}$$

where Pr is the Prandtl number of the cavity fill gas.

A Nusselt number of unity implies heat transfer by fluid conduction only, while a higher than unity Nusselt number is due to the so-called "Rayleigh" effect which monotonically increases with increasing Rayleigh number. Nusselt numbers applicable to the peripheral voids in helium-filled PWR and BWR fueled HI-STAR 100 MPCs in vertically-oriented casks are provided in Table 4.4.1.

The above discussion applies to the peripheral voids in helium-filled MPCs in vertically-oriented casks. An analogous discussion for the peripheral voids in helium-filled MPCs in horizontally-oriented casks is presented in Subparagraph 3.4.1.1.5 of the HI-STAR 100 transport SAR, which presents corresponding Nusselt numbers in its Table 3.4.1.

4.4.1.1.6 Effective Conductivity of Multilayered Intermediate Shell Region

Fabrication of the multi-layered overpack shell is discussed in Section 1.2 which explains how an interfacial contact between successive layers from the fabrication process is ensured. In the thermal analysis, each intermediate shell metal-to-metal interface presents an additional resistance to heat transport. The contact resistance arises from microscopic pockets of air entrapped between surface irregularities of the contacting surfaces. Since air is a relatively poor conductor of heat, this results in a reduction in the ability to transport heat across the interface compared to that of the base metal. Interfacial contact conductance depends upon three principal factors, namely: (i) base material conductivity, (ii) interfacial contact pressure, and (iii) surface finish. Rohsenow and Hartnett [4.2.2] have reported results from experimental studies of contact conductance across air entrapped stainless steel surfaces with a typical 100 μ-inch (2.54 μm) surface finish. A minimum contact conductance of 350 Btu/ft^2-hr-°F (606 W/m-K) is determined from extrapolation of Rohsenow, et al. data to zero contact pressure.

Thermal conductivity of carbon steel is about three times that of stainless steel. Thus, the choice of carbon steel as base material in a multi-layered construction significantly improves heat transport across interfaces. The fabrication process, as discussed in Section 1.2, guarantees significant interfacial contact. Contact conductance values extrapolated to zero contact pressure are therefore conservative. The surface finish of the hot-rolled carbon steel plate stock is generally in the range of 250-1000 μ-inch (6.4-25.4 μm) [4.2.1]. The process of forming hot-rolled flat plate stock to cylindrical shapes to form the intermediate shells will result in additional smoothening of the surfaces (from the large surface pressures exerted by the hardened roller faces which flatten out any surface irregularities).

In the HI-STAR 100 thermal analysis, a conservatively bounding interfacial contact conductance value is determined using the following assumptions:

1. No credit is taken for higher base metal conductivity (carbon versus stainless steel).

2. No credit is taken for interfacial contact pressure.

3. No credit is taken for a smooth surface finish resulting from rolling of hot-rolled plate stock to cylindrical shapes.

4. Contact conductance is based on a uniform 2000 μ-inch (50.8 μm) (1000 μ-inch (25.4 μm) for each surface condition) interfacial air gap at all interfaces.

5. No credit for radiation heat exchange across this hypothetical inter-surface air gap.

6. Bounding low thermal conductivity at 200°F (93°C).

These assumptions guarantee a conservative assessment of heat dissipation characteristics of the multi-layered intermediate shell region. The resistance of the five carbon steel layers along with the associated interfacial resistances are combined as resistances in series to determine an effective conductivity of this region leading to the following relationship:

$$K_i = r_o \ell n\left(\frac{r_5}{r_o}\right)\left[\sum_{i=1}^{5}\frac{\delta}{K_{air}}\frac{r_o}{r_i}+\frac{r_o \ell n \frac{r_5}{r_o}}{K_{cst}}\right]^{-1}$$

where (in conventional U.S. units):

K_i	=	effective intermediate shell region thermal conductivity
r_o	=	inside radius of inner intermediate shell
r_i	=	outer radius of i^{th} intermediate shell
δ	=	interfacial air gap (2000 μ-inch (50.8 μm))
K_{air}	=	air thermal conductivity
K_{cst}	=	carbon steel thermal conductivity

4.4.1.1.7 Heat Rejection from Overpack Exterior Surfaces

Jacob and Hawkins [4.2.9] recommend the following correlations for natural convection heat transfer to air from heated vertical cylinders or horizontal flat surfaces:

Turbulent range:

$$h = 0.19\,(\Delta T)^{1/3}\ (Vertical, GrPr > 10^9)$$

$$h = 0.22\,(\Delta T)^{1/3}\ (Horizontal, GrPr > 10^7)$$

$$(in\ conventional\ U.S.\ units)$$

Laminar Range:

$$h = 0.29 \left(\frac{\Delta T}{L} \right)^{1/4} (Vertical\ GrPr < 10^9)$$

$$h = 0.27 \left(\frac{\Delta T}{L} \right)^{1/4} (Horizontal\ GrPr < 2x10^7)$$

$$(in\ conventional\ U.S.\ units)$$

where ΔT is the temperature differential between the overpack surface and ambient air. The length scale L is the overpack height for vertical surfaces or the overpack diameter for the top horizontal surface. Noting that GrPr is expressed as $L^3 \Delta TZ$, where Z (from Table 4.2.7) is at least $2.6x10^5$ at a conservatively high upper bound overpack exterior air film temperature of 340°F (171°C), it is apparent that the turbulent condition is always satisfied for ΔT in excess of a small fraction of 1°F (0.56°C). Under turbulent conditions, the more conservative heat transfer correlation for vertical cylindrical surfaces (i.e., $h = 0.19\ \Delta T^{1/3}$) is applied for thermal analysis to all exposed overpack surfaces.

Jacob and Hawkins [4.2.9] also recommend correlations for natural convection heat transfer to air from heated horizontal cylinders and vertical flat surfaces (as described in Subparagraph 3.4.1.1.7 of the HI-STAR 100 transport SAR).

Including both natural convection and thermal radiation heat loss from the overpack outer surfaces, the following relationship for surface heat flux is developed for vertically-oriented casks:

$$q_s = 0.19\ (T_s - T_A)^{4/3} + \sigma\ \varepsilon\ F_{1,A}\ [(T_s + 460)^4 - (T_A + 460)^4]$$

where:

T_s, T_A	=	surface, ambient temperatures (°F)
q_s	=	surface heat flux (Btu/ft^2-hr)
ε	=	surface emissivity
$F_{1,A}$	=	view factor between surface and air
σ	=	Stefan Boltzmann constant ($0.1714x10^{-8}$ Btu/ft^2-hr-°R^4)

An analogous relationship is developed for horizontally-oriented casks in Subparagraph 3.4.1.1.7 of the HI-STAR 100 transport SAR.

In order to determine the view factor for vertical overpack outside surfaces, an ANSYS [4.1.1] finite-element based radiation heat transfer model is developed. The model geometry is based on a HI-STAR 100 System array layout depicted schematically in Figure 1.4.1. The design basis HI-STAR 100 System ISFSI storage square layout pitch is provided in Section 1.4. The ANSYS model developed is shown in Figure 4.4.5. In this figure, a center HI-STAR 100 System cask is shown surrounded by two rows of casks on all sides. The ANSYS solution determines view factors between this most adversely located system in the middle with all other neighboring casks. A sum of all these individual blockages gives the total blockage factor. Thus, the view factor $F_{1,A}$ between this most adversely affected HI-STAR 100 System and outside air is determined by the following relationship:

$$F_{1,A} = 1 - \sum_{K} F_{1,K}$$

where $F_{1,K}$ is the view factor between HI-STAR 100 System 1 and a neighboring system K. This factor is determined by a series of ANSYS solutions as a function of ISFSI cask array pitch, and the results are shown in Figure 4.4.6.

For the surfaces of a horizontally-oriented cask, blocking factors due to adjacent casks are determined in the lateral direction (i.e., blocking by casks in parallel rows) and the axial direction (i.e., blocking of the lid surface of a cask by the base of an adjacent cask in the same row). View factors are determined for each of these surfaces (cask side = lateral; cask lid = axial) using the same equation for F1,A above. The design-basis spacing from Section 1.4 is evaluated.

4.4.1.1.8 Determination of Solar Heat Input

The intensity of solar radiation incident on an exposed surface depends on a number of time varying terms. The solar heat flux strongly depends upon the time of the day as well as on latitude and day of the year. Also, the presence of clouds and other atmospheric conditions (dust, haze, etc.) can significantly attenuate solar intensity levels. Rapp [4.4.2] has discussed the influence of such factors in considerable detail.

Consistent with the guidelines in NUREG-1536 [4.1.3], solar input to the exposed surfaces of the overpack is determined based on 12-hour insolation levels recommended in 10CFR71 (averaged over a 24-hour period) and applied to the most adversely located cask after accounting for partial blockage of incident solar radiation on the lateral surfaces of the cask by surrounding casks. The blocking factor is identical to the radiative blocking considered for cooling of outside surfaces to the ambient environment. This is conservative compared to the case of an isolated cask with significantly improved radiative cooling and higher insolation levels because the cask is emitting much more heat than the insolation heat input. The imposed steady insolation level for the exposed top lid is based on a view factor equal to unity. The solar absorptivity of all exposed cask surfaces is assumed to be a conservatively bounding value of unity.

4.4.1.1.9 Effective Thermal Conductivity of Holtite Neutron Shielding Region

In order to minimize heat transfer resistance limitations due to the poor thermal conductivity of the Holtite-A neutron shield material, a large number of thick radial channels of high strength and conductivity carbon steel material are embedded in the neutron shield region. The legs of the radial channels form highly conducting heat transfer paths for efficient heat removal. Each channel leg is welded to the outside surface of the outermost intermediate shell. Enclosure shell panels are welded to the radial channels to form the external wall of the overpack, and thus provide a continuous path for heat removal to the ambient environment.

The effective thermal conductivity of the composite neutron shield and the network of radial channel legs is determined by combining the heat transfer resistance of individual components in a parallel network. In determining the heat transfer capability of this region to the outside ambient environment for normal long-term storage conditions, <u>no credit is taken for conduction through the neutron shielding material.</u> Thus, heat transport from the outer intermediate shell surface to the overpack outer shell is conservatively based on heat transfer through the carbon steel radial connectors alone. Thermal conductivity of the parallel neutron shield and radial channel leg region is given by the following formula:

$$K_{ne} = \frac{K_r\, N_r\, t_r\, \ell n\left(\frac{r_B}{r_A}\right)}{2\pi\, \ell_R} + \frac{K_{ns}\, N_r\, t_{ns}\, \ell n\left(\frac{r_B}{r_A}\right)}{2\pi\, \ell_R}$$

where (in consistent U.S. units):

K_{ne}	=	effective thermal conductivity of Holtite region
r_A	=	inner radius of neutron shielding
r_B	=	outer radius of neutron shielding
K_r	=	effective thermal conductivity of carbon steel radial channel leg
N_r	=	total number of radial channel legs (also equal to number of neutron shield sections)
t_r	=	minimum (nominal) thickness of each radial channel leg
ℓ_R	=	effective radial heat transport length through radial channel leg
K_{ns}	=	neutron shield thermal conductivity
t_{ns}	=	neutron shield thickness (between two radial channel legs)

The radial channel-to-outer intermediate shell surface weld thickness is equal to half the plate thickness. The additional weld resistance is accounted for by reducing the plate thickness in the weld region for a short radial span equal to the weld size. As a result, the conductivity of the radial carbon steel connectors based on full thickness for the entire radial span is reduced. Figure 4.4.7 depicts a resistance network developed to combine the neutron shield and radial connectors resistances to determine an effective conductivity of the neutron shield region. Note that in the resistance network analogy, only the annulus region between overpack outer enclosure inner surface and intermediate shells outer surface is considered in this analysis. The effective thermal conductivity of the neutron shield/radial channel leg region is provided in Table 4.4.8.

4.4.1.1.10 Effective Thermal Conductivity of Flexible MPC Basket-to-Shell Aluminum Heat Conduction Elements

As shown in HI-STAR 100 System MPC drawings in Section 1.5, flexible full-length heat conduction elements fabricated from thin aluminum alloy 1100 sheet metal are inserted in the large MPC basket-to-shell gaps for certain configurations to provide uninterrupted metal pathways to transport heat from the basket to the MPC shell. Due to the high thermal conductivity of aluminum alloy 1100 (about 15 times that of Alloy X), a significant rate of heat transfer is possible along thin flexible plates. Flexibility of the heat conduction elements is an important asset to enable a snug fit in the confined spaces and for ease of installation. Figure 4.4.13 shows the mathematical idealization of a typical conduction element inserted in a basket periphery panel-to-MPC shell space. The aluminum heat conduction element is shown to cover the MPC basket Alloy X peripheral panel and MPC shells (Regions I and III depicted in Figure 4.4.13) surfaces along the full-length of the basket except for isolated locations where fit-up or interference with other parts precludes complete basket coverage. Heat transport to and from the aluminum heat conduction element is conservatively postulated to occur across a thin helium gap as shown in the figure (i.e., no credit is taken for aluminum heat conduction element to Alloy X metal-to-metal contact). Aluminum surfaces inside the hollow region are sandblasted prior to fabrication to result in a rough surface finish which has a significantly higher emissivity compared to smooth surfaces of rolled aluminum. The untreated aluminum surfaces directly facing Alloy X panels have a smooth finish to minimize contact resistance.

Net heat transfer resistance from the hot basket periphery panel to the relatively cooler MPC shell along the aluminum heat conduction element pathway is a sum of three individual resistances in regions labeled I, II, and III as shown in Figure 4.4.13. In Region I, heat is transported from the basket to the aluminum heat conduction element surface directly facing the basket panel across a thin helium resistance gap. Longitudinal transport of heat (in the z direction) in the aluminum plate (in Region I) will result in an axially non-uniform temperature distribution. Longitudinal one-dimensional heat transfer in the Region I aluminum plate was analytically formulated to result in the following ordinary differential equation for the non-uniform temperature distribution:

$$t \; K_{A\ell} \frac{\partial^2 T}{\partial z^2} = - \frac{K_{He}}{h} (T_h - T) \qquad \text{(Equation a)}$$

Boundary Conditions

$$\frac{\partial T}{\partial z} = 0 \text{ at } z = 0$$
$$T = T_h' \text{ at } z = P \qquad \text{(Equation b)}$$

where (see Figure 4.4.13):

T(z) = non-uniform aluminum metal temperature distribution

t = heat conduction element thickness

K_{Al} = heat conduction element conductivity

K_{He} = helium conductivity

h = helium gap thickness

T_h = hot basket temperature

T_h' = heat conduction element Region I boundary temperature at z = P

P = heat conduction element Region I length

W = conduction element Region II length

Solution of this ordinary differential equation subject to the imposed boundary condition is:

$$(T_h - T) = (T_h - T_h') \left[\frac{e^{\frac{z}{\sqrt{\alpha}}} + e^{-\frac{z}{\sqrt{\alpha}}}}{e^{\frac{P}{\sqrt{\alpha}}} + e^{-\frac{P}{\sqrt{\alpha}}}} \right] \quad \text{(Equation c)}$$

where α is a dimensional parameter equal to ($h \times t \times K_{Al}/K_{he}$). The net heat transfer (Q_I) across the Region I helium gap can be determined by the following integrated heat flux to a heat conduction element of length L as:

$$Q_I = \int_0^P \frac{K_{He}}{h} (T_h - T)\ (L)\ dz \quad \text{(Equation d)}$$

Substituting the analytical temperature distribution result obtained in Equation c, the following expression for net heat transfer is obtained:

$$Q_I = \frac{K_{He}\ L\ \sqrt{\alpha}}{h} \left(1 - \frac{1}{e^{\frac{P}{\sqrt{\alpha}}} + e^{-\frac{P}{\sqrt{\alpha}}}} \right) (T_h - T_h') \quad \text{(Equation e)}$$

Based on this result, an expression for Region I resistance is obtained as shown below:

$$R_I = \frac{T_h - T_{h'}}{Q_I} = \frac{h}{K_{He}\ L\ \sqrt{\alpha}} \left(1 - \frac{1}{e^{\frac{P}{\sqrt{\alpha}}} + e^{-\frac{P}{\sqrt{\alpha}}}} \right)^{-1} \quad \text{(Equation f)}$$

The Region II resistance expression can be developed from the following net heat transfer equation in the vertical leg of the conduction element as shown below:

$$Q_{II} = \frac{K_{Al}\ L\ t}{W} (T_h' - T_c') \quad \text{(Equation g)}$$

$$R_{II} = \frac{T_h' - T_c'}{Q_{II}} = \frac{W}{K_{Al}\ L\ t} \qquad \text{(Equation h)}$$

Similarly, a Region III resistance expression can be analytically determined as shown below:

$$R_{III} = \frac{(T_c' - T_c)}{Q_{III}} = \frac{h}{K_{He}\ L\ \sqrt{\alpha}} \left(1 - \frac{1}{e^{\frac{P}{\sqrt{\alpha}}} + e^{-\frac{P}{\sqrt{\alpha}}}} \right)^{-1} \qquad \text{(Equation i)}$$

This completes the analysis for the total thermal resistance attributable to the heat conduction elements, equal to the sum of the three individual resistances. The total heat conduction element resistance is smeared across the basket-to-MPC shell region as an effective uniform annular gap conductivity (see Figure 4.4.2). We note that heat transport along the conduction elements is an independent conduction path in parallel with conduction and radiation mechanisms in the large helium gaps. Helium conduction and radiation in the MPC basket-to-MPC shell peripheral gaps is accounted for separately in the ANSYS models for the MPCs, described earlier. Therefore, the total MPC basket-to-MPC shell peripheral gaps conductivity will be the sum of the heat conduction elements effective conductivity and the helium gap conduction-radiation effective conductivity.

4.4.1.1.11 FLUENT Model for HI-STAR 100 Temperature Field Computation

In the preceding subparagraphs, a series of analytical and numerical models to define the thermal characteristics of the various elements of the HI-STAR 100 System are presented. The thermal modeling begins with the replacement of the SNF cross section and surrounding fuel cell space with a solid region with an equivalent conductivity. Since radiation is an important constituent of the heat transfer process in the SNF/storage cell space, and the rate of radiation heat transfer is a strong function of the surface temperatures, it is necessary to treat the equivalent region conductivity as a function of temperature. Because of the relatively large range of temperatures in a loaded HI-STAR 100 System under the design basis heat loads, the effects of variation in the thermal conductivity of materials with temperature throughout the system model are included. The presence of significant radiation effects in the storage cell spaces adds to the imperative to treat the equivalent storage cell lamina conductivity as temperature-dependent.

FLUENT finite volume simulations have been performed to establish the equivalent thermal conductivity as a function of temperature for the limiting (thermally most resistive) BWR and PWR spent fuel types. Utilizing the limiting SNF (established through a simplified analytical process for comparing conductivities) ensures that the numerical idealization for the fuel space effective conductivity is conservative for all non-limiting fuel types.

Having replaced the fuel spaces by solid square blocks with temperature-dependent conductivity essentially renders the basket into a non-homogeneous three-dimensional solid where the non-homogeneity is introduced by the honeycomb basket structure. The basket panels themselves are a composite of Alloy X cell wall, neutron absorber, and Alloy X sheathing metal. A conservative approach to replace this composite section with an equivalent "solid wall" was described earlier.

In the next step, a planar section of the MPC is considered. The MPC contains a non-symmetric basket lamina wherein the equivalent fuel spaces are separated by the "equivalent" solid metal walls. The space between the basket and the MPC, called the peripheral gap, is filled with helium gas and aluminum heat conduction elements (shown in the MPC drawings in Section 1.5). The equivalent thermal conductivity of the MPC section is computed using a finite element procedure on ANSYS. To the "helium conduction-radiation" based peripheral gap conductivity, the effective conductivity of the aluminum conduction elements is added to obtain a combined peripheral gap effective conductivity. At this stage in the thermal analysis, the SNF/basket/MPC assemblage has been replaced with a two-zone (Figure 4.4.2) cylindrical solid whose thermal conductivity is a strong function of temperature.

The idealization for the overpack is considerably more straightforward. The overpack is radially symmetric except for the neutron absorber (Holtite-A) region (Figure 4.4.7). The procedure to replace the multiple shell layers, Holtite-A and radial connectors with an equivalent solid utilizes classical heat conduction analogies, as discussed in Subparagraphs 4.4.1.1.6 and 4.4.1.1.9.

In the final step of the analysis, the equivalent two-zone MPC cylinder, equivalent overpack shell, top and bottom plates, and ISFSI pad are assembled into a comprehensive finite volume model. A cross section of the axisymmetric model implemented on FLUENT for vertically-oriented casks is shown in Figure 4.4.15. A summary of the essential features of this model is presented in the following:

- The overpack shell is represented by 840 axisymmetric elements.

- The overpack bottom plate and bolted closure plate are modeled by 312 axisymmetric elements.

- The two-zone MPC "solid" (including the baseplate, lid and shell) is represented by 1188 axisymmetric elements.

- The ISFSI pad is conservatively modeled as a thermal resistance from a 36" thick concrete cylinder whose bottom surface is at 60°F. The portion of the concrete outside the footprint of the cask is conservatively omitted from the model.

- The space between the MPC and the overpack interior inner surface contains helium.

- Heat input due to insolation is applied to the top surface and the cylindrical surface of the overpack.

- The heat generation in the MPC is assumed to be uniform in each horizontal plane, but to vary in the axial direction to correspond to the axial power distribution listed in Table 2.1.8.

- The most disadvantageously placed cask in a HI-STAR cask array (i.e., the one subjected to maximum radiative blockage (see Subsection 4.4.1.1.7), is modeled.

The emissivity applied to the external surfaces of the HI-STAR model accounts for radiation-blockage of the outer enclosure surface and no blockage for the overpack closure plate top surface.

The finite element model constructed in this manner will produce an axisymmetric temperature distribution. The peak temperature will occur at the centerline and is expected to occur at the axial location of peak heat generation. As we will see later, the results from the finite volume solution bear out these observations.

A cross section of the 3-D axisymmetric model implemented on FLUENT for horizontally-oriented casks is shown in Figure 3.4.14 of the HI-STAR 100 transport SAR [4.0.1]. See Subparagraph 3.4.1.1.12 of that document for a description of the model for horizontally-oriented casks.

4.4.1.1.12 MPC Temperature Distribution Under Drying Conditions

The initial loading of SNF in the MPC requires that the water within the MPC be drained and replaced with helium. This operation on the HI-STAR MPCs will be carried out using either a conventional vacuum drying approach or using a Forced Helium Dehydrator (FHD). In this method, removal of the last traces of residual moisture from the MPC cavity is accomplished by evacuating the MPC for a short time after draining the MPC.

Vacuum Drying

Prior to the start of the MPC draining operation, both the overpack annulus and the MPC are full of water. The presence of water in the MPC ensures that the fuel cladding temperatures are lower than design basis limits by large margins. As the heat generating active fuel length is uncovered during the draining operation, the fuel and basket mass will undergo a gradual heat up from the initially cold conditions when the heated surfaces were submerged under water.

Thermal analysis of the MPC basket with MBF for bounding design basis decay heat loads is performed on the ANSYS finite element code. The ANSYS model is constructed to evaluate the heat rejection ability of the basket under evacuated conditions. The vacuum condition effective fuel assembly conductivity is determined by procedures discussed earlier (Subparagraph 4.4.1.1.2) after setting the thermal conductivity of the gaseous medium to a small fraction (one part in one thousand) of helium conductivity in the fuel assembly finite element model. Basket periphery-to-MPC shell heat transfer occurs through conduction and radiation. During draining and vacuum drying operations, the overpack annulus is required to be kept filled with water. Thus, the MPC thermal analysis problem is formulated with cooling of the MPC shell with water, which under worst case conditions would be slightly higher than its normal boiling temperature at the bottom of the overpack annulus. Results of vacuum condition analyses are provided in Paragraph 4.4.2.2.

Forced Helium Dehydration

To reduce moisture to trace levels in the MPC using a Forced Helium Dehydration (FHD) system, a closed loop system consisting of a condenser, a demoisturizer, a compressor, and a pre-heater is utilized to extract moisture from the MPC cavity through repeated displacement of its contained helium, accompanied by vigorous flow turbulation. Appendix 4.A contains detailed discussion of the design and operation criteria for the FHD system.

The FHD system provides concurrent fuel cooling during the moisture removal process through forced convective heat transfer. The attendant forced convection-aided heat transfer occurring during operation of the FHD system ensures that the fuel cladding temperature will remain below the applicable peak cladding temperature limit for normal conditions (752°F or 400°C) for all combinations of SNF type, burnup, decay heat, and cooling time. Because the FHD operation induces a state of forced convection heat transfer in the MPC, (in contrast to the quiescent mode of natural convection in storage), it is readily concluded that the peak fuel cladding temperature under the latter condition will be greater than that during the FHD operation phase. In the event that the FHD system malfunctions, the forced convection state will degenerate to natural convection, which corresponds to the conditions of normal storage. As a result, the peak fuel cladding temperatures will approximate the values reached during normal conditions as described elsewhere in this chapter.

4.4.1.1.13 Deleted

4.4.1.1.14 Maximum Time Limit During Wet Transfer

In accordance with NUREG-1536, water inside the MPC cavity during wet transfer operations is not permitted to boil in the HI-STAR 100 System. Consequently, uncontrolled pressures in the de-watering, purging, and recharging system which may result from two-phase condition, are completely avoided. This requirement is accomplished by imposing a limit on the maximum allowable time duration for fuel to be submerged in water after a loaded HI-STAR cask is removed from the pool and prior to the start of MPC cavity drying operations.

When the HI-STAR overpack and the loaded MPC under water-flooded conditions are removed from the pool, the combined mass of the water, the fuel, the MPC, and the HI-STAR will absorb the decay heat emitted by the fuel assemblies. This results in a slow temperature rise of the entire system with time, starting from an initial temperature of the contents. The rate of temperature rise is limited by the thermal inertia of the HI-STAR system. To enable a bounding heat-up rate determination for the HI-STAR system, the following conservative assumptions are imposed:

i. Heat loss by natural convection and radiation from the exposed HI-STAR surfaces to the pool building ambient air is neglected (i.e., an adiabatic temperature rise calculation is performed).

ii. Design Basis maximum decay heat input from the loaded fuel assemblies is imposed on the HI-STAR system.

iii. The smallest of the minimum MPC cavity-free volumes among the three MPC types is considered for flooded water mass determination.

iv. Fifty percent of the water mass in the MPC cavity is credited towards water thermal inertia evaluation.

Table 4.4.20 summarizes the weights and thermal inertias of several components in the loaded HI-STAR system. The rate of temperature rise of the HI-STAR and its contents during an adiabatic heat-up is governed by the following equation:

$$\frac{dT}{dt} = \frac{Q}{C_h}$$

where:

Q = decay heat load (Btu/hr)
[equal to Design Basis maximum (among the three MPC types) 20 kW (i.e., 68,260 Btu/hr)]

C_h = combined thermal inertia of the loaded HI-STAR system (Btu/°F)

T = temperature of the contents (°F)

t = time after HI-STAR system is removed from the pool (hr)

A bounding heat-up rate for the HI-STAR system contents is determined to be equal to 2.19°F/hr (1.22°C/hr). From this adiabatic rate of temperature rise estimate, the maximum allowable time duration (t_{max}) for fuel to be submerged in water is determined as follows:

$$t_{max} = \frac{T_{boil} - T_{initial}}{(dT/dt)}$$

where:

T_{boil} = boiling temperature of water
(equal to 212ºF (100°C) at the water surface in the MPC cavity)

$T_{initial}$ = initial temperature of the HI-STAR contents when removed from the pool

Table 4.4.21 provides a summary of t_{max} at several initial HI-STAR contents temperatures.

As set forth in the HI-STAR 100 operating procedures, in the unlikely event where the maximum allowable time provided in Table 4.4.21 is found to be insufficient to complete all wet transfer operations, a forced water circulation shall be initiated and maintained to remove the decay heat from the MPC cavity. In this case, relatively cooler water will enter via the MPC lid drain port connection and heated water will exit from the vent port. The minimum water flow rate required to maintain the MPC cavity water temperature below boiling with an adequate subcooling margin is determined as follows:

$$M_W = \frac{Q}{C_{pw}\ (T_{max} - T_{in})}$$

where:

M_W = minimum water flow rate (lb/hr)

C_{pw} = water heat capacity (Btu/lb-ºF)

T_{max} = maximum MPC cavity water mass temperature

T_{in} = temperature of water supply to MPC

With the MPC cavity water temperature limited to 150ºF (66°C), MPC inlet water maximum temperature equal to 125ºF (52°C) and at the design basis maximum heat load, the water flow rate is determined to be 2,731 lb/hr (1241 kg/hr) or 5.5 gpm (1249 L/hr).

4.4.1.1.15 Cask Cooldown and Reflood Analysis During Fuel Unloading Operation

NUREG-1536 requires an evaluation of cask cooldown and reflood procedures to support fuel unloading from a dry condition. Past industry experience generally supports cooldown of cask internals and fuel from hot storage conditions by direct water quenching. Direct MPC cooldown is effectuated by introducing water through the lid drain line. From the drain line, water enters the MPC cavity near the MPC baseplate. Steam produced during the direct quenching process will be vented from the MPC cavity through the lid vent port. To maximize venting capacity, both vent port RVOA connections must remain open for the duration of the fuel unloading operations. As direct water quenching of hot fuel results in steam generation, it is necessary to limit the rate of water addition to avoid MPC overpressurization. For example, steam flow calculations using bounding assumptions (100% steam production and MPC at design pressure) show that the MPC is adequately protected up to a reflood rate of 3715 lb/hr (1689 kg/hr). Limiting the water reflood rate to this amount or less would prevent exceeding the MPC design pressure.

During direct reflood operations the fuel cladding is subject to high temperature gradients and concomitant thermal stresses. The integrity of fuel under direct quenching is evaluated in a generic manner in the HI-STORM FW SAR (Docket No. 72-1032, Ref. [4.4.7]). To define a bounding scenario at time t = 0 sec, a uniformly bounding temperature throughout the entire fuel rod is set at 752ºF (400ºC), which is the temperature limit of fuel cladding. At time t = 0.1 sec, a reasonably bounding 80ºF (27ºC) quench water temperature is assigned to the lower half of the fuel rod to simulate a thermal shock with a large step change in the cladding temperature. The resulting transient stress and strain distributions in the fuel rod are evaluated with finite element ANSYS models. The results show that the maximum stress and strain values remain within the elastic range and remain well within failure strain limit (a factor of 6 against failure strain). This safety analysis documented in Subparagraph 3.4.4.1.11 of the HI-STORM FW FSAR provides the assurance that the MPC reflood event will not cause a breach of fuel cladding.

4.4.1.1.16 HI-STAR Temperature Field With Low Emitting Fuel

The HI-STAR 100 thermal evaluations for BWR fuel are divided in two groups of fuel assemblies proposed for storage in MPC-68. These groups are classified as Low Heat Emitting (LHE) fuel assemblies and Design Basis (DB) fuel assemblies. The LHE group of fuel assemblies are characterized by low burnup, long cooling time, and short active fuel lengths. Consequently, their heat loads are dwarfed by the DB group of fuel assemblies. The Dresden-1 (6x6 and 8x8), Quad+, and Humboldt Bay (7x7 and 6x6) fuel characteristics warrant their classification as LHE fuel. These characteristics, including burnup and cooling time limits imposed on this class of fuel, are presented in Table 2.1.6. This fuel (except Quad$^+$) is permitted to be loaded when encased in Damaged Fuel Containers (DFCs). As a result of interruption of radiation heat exchange between the fuel assembly and the fuel basket by the DFC boundary, this loading configuration is bounding for thermal evaluation. In Subparagraph 4.4.1.1.2, two canister designs for encasing LHE fuel are evaluated – a previously approved Holtec Design (Holtec Drawing-1783) and an existing canister in which some of the Dresden-1 fuel is currently stored (Transnuclear D-1 canister). The most resistive fuel assembly determined by analytical evaluation is considered for thermal evaluation (see Table 4.4.6). The MPC-68 basket effective conductivity, loaded with the most resistive fuel assembly from the LHE group of fuel (encased in a canister) is provided in Table 4.4.7. To this basket, LHE decay heat load is applied and a HI-STAR 100 System temperature field obtained. The low heat load burden limits the initial peak cladding temperature to 595°F (313°C) which is substantially below the long-term temperature limit.

A thoria rod canister designed to hold a maximum of 20 fuel rods arrayed in a 5x4 configuration is currently stored at the Dresden-1 spent fuel pool. The fuel rods contain a mixture of enriched UO_2 and Thorium Oxide in the fuel pellets. The fuel rods were originally constituted as part of an 8x8 fuel assembly and used in the second and third cycle of Dresden-1 operation. The maximum fuel burnup of these rods is quite low (~14,400 MWD/MTU). The thoria rod canister internal design is a honeycomb structure formed from 12 gage stainless steel plates. The rods are loaded in individual square cells and are isolated from each other by the cell walls. The few number of rods (18 per assembly) and very low burnup of fuel stored in these Dresden-1 canisters render them as miniscule sources of decay heat. The canister all-metal internal honeycomb construction serves as an additional means of heat dissipation in the fuel cell space. In accordance with the loading requirements imposed in the Technical Specifications, thoria fuel shall be loaded toward the basket periphery (i.e., away from the hot central core of the fuel basket). All these considerations provide ample assurance that these fuel rods will be stored in a benign thermal environment and therefore remain protected during long-term storage.

4.4.1.2 Test Model

A detailed analytical model for thermal design of the HI-STAR 100 System was developed using the FLUENT CFD code and the industry standard ANSYS modeling package, as discussed in

Subparagraph 4.4.1.1. As discussed throughout this chapter and specifically in Subsection 4.4.6, the analysis incorporates significant conservatisms so as to predict the fuel cladding temperature with considerable margins. Furthermore, compliance with specified limits of operation is demonstrated with adequate margins. In view of these considerations, the HI-STAR 100 System thermal design complies with the thermal criteria set forth in the design basis (Sections 2.1 and 2.2) for long-term storage under normal conditions. Additional experimental verification of the thermal design is therefore not required.

4.4.2 Maximum Temperatures

4.4.2.1 Maximum Temperatures Under Normal Storage Conditions

The MPC basket designs developed for the HI-STAR 100 System have been analyzed to determine the temperature distribution under long-term normal storage conditions. The MPC baskets are considered to be loaded at design basis maximum heat loads with PWR or BWR fuel assemblies, as appropriate. The systems are considered to be arranged in an ISFSI array and subjected to design basis normal ambient conditions with insolation.

Applying the radiative blocking factor applicable for the worst case cask location, converged temperature contours are shown in Figures 4.4.17 and 4.4.18 for the MPC-24, and MPC-68 basket designs in a vertical orientation. The temperatures in these two figures are in degrees Kelvin. Analogous figures for horizontal orientation storage of the MPC-24 and MPC-68 are presented in Figures 3.4.16 and 3.4.17 of the HI-STAR 100 transportation SAR. The calculated temperatures presented in this chapter are based on an array of analyses that incorporate many conservatisms. As such, the calculated temperatures are upper bound values which would exceed actual temperatures.

The maximum fuel clad temperatures for zircaloy clad fuel assemblies in vertically-oriented casks are listed in Tables 4.4.10 and 4.4.11, which also summarize maximum calculated temperatures in different parts of the HI-STAR 100 System. Figures 4.4.21 and 4.4.22 show the axial temperature variation of the hottest fuel rod in the MPC-24 and MPC-68 basket designs, respectively. Figures 4.4.24 and 4.4.25 show the radial temperature profile in the MPC-24 and MPC-68 basket designs, respectively, in the horizontal plane where maximum fuel cladding temperature is indicated.

Applying the radiative blocking factor applicable for the worst case cask location, the maximum fuel clad temperature for zircaloy clad fuel assemblies in an MPC-32 in a vertically-oriented HI-STAR 100 System is listed in Table 4.4.12, which also summarizes maximum calculated temperatures in different parts of the HI-STAR 100 System.

Comparing the fuel cladding temperatures and the temperatures of the bounding confinement boundary component (the MPC shell) in Tables 4.4.11 and 4.4.12, the MPC-68 results in higher temperatures than the MPC-32. It is therefore concluded that the MPC-68 bounds the MPC-32 when both are in vertically-oriented casks.

The maximum fuel clad temperatures for zircaloy clad fuel assemblies in horizontally-oriented casks are listed in Tables 3.4.10 and 3.4.11 of the HI-STAR 100 transportation SAR [4.0.1], which also summarize maximum calculated temperatures in different parts of the HI-STAR 100 System. Comparing the fuel cladding temperatures and the temperatures of the bounding confinement boundary component (the MPC shell) in these two SAR tables, the MPC-68 results in higher temperatures than the MPC-32. It is therefore concluded that the MPC-68 bounds the MPC-32 when both are in horizontally-oriented casks.

The horizontally-oriented casks are supported by structures that holds the casks and prevents their movement during design-basis events. This structure covers the cask surface at two locations along its length as shown in Figure 1.2.13. The transportation thermal performance incorporated by reference from the HI-STAR 100 SAR is compared to the normal storage condition with due consideration of the support structure, as shown in Table 4.4.26. As this table shows, the thermal performance of the cask under horizontal storage conditions while mounted in the support structure is essentially the same as or somewhat superior to the horizontal transport conditions incorporated by reference. More critically, it is shown that all fission product boundary temperatures and internal pressures remain below their respective limits. It is concluded, therefore, that the safety analysis for the transport condition does conservatively represent the thermal performance under the horizontal normal storage condition.

Similarly, the effects of array blocking for an array of horizontally-oriented casks is evaluated in the same manner as described in Subparagraph 4.4.1.1.7 for the normal storage condition and compared to the transportation thermal performance incorporated by reference from the HI-STAR 100 SAR in Table 4.4.27. As this table shows, for the cask under horizontal storage conditions arrayed at the proposed pitches, all fission product boundary temperatures and internal pressures remain below their respective limits. The increase in the components' temperatures is minor. It is concluded, therefore, that the proposed separation between the horizontally stored casks is suitable for horizontal deployment of HI-STAR 100 Systems.

As discussed in Subsection 4.4.1.1.1, the thermal analysis is performed using a submodeling process where the results of an analysis on an individual component are incorporated into the analysis of a larger set of components. Specifically, the submodeling process yields directly computed fuel temperatures from which fuel basket temperatures are indirectly calculated. This modeling process differs from previous analytical approaches wherein the basket temperatures were evaluated first and then a basket-to-cladding temperature difference calculation by Wooten-Epstein or other means provided a basis for cladding temperatures. Subsection 4.4.1.1.2 describes the calculation of an effective fuel assembly thermal conductivity for an equivalent homogenous region. It is important to note that the result of this analysis is a function for thermal conductivity versus temperature. This function for fuel thermal conductivity is then input to the fuel basket effective thermal conductivity calculation described in Subsection 4.4.1.1.4. This calculation uses a finite-element methodology, wherein each fuel cell region containing multiple finite-elements has temperature varying thermal conductivity properties. The resultant temperature varying fuel basket thermal conductivity computed by this basket-fuel composite model is then input to the fuel basket region of the FLUENT cask model.

Because the FLUENT cask model incorporates the results of the fuel basket submodel, which in turn incorporates the fuel assembly submodel, the peak temperature reported from the FLUENT model is the peak temperature in any component. In a dry storage cask, the hottest components are the fuel assemblies. It should be noted that, because the fuel assembly models described in Subsection 4.4.1.1.2 include the fuel pellets, the FLUENT calculated peak temperatures reported in Tables 4.4.10 and 4.4.11 for vertically-oriented casks and in the HI-STAR 100 transport SAR for horizontally-oriented casks are actually peak pellet centerline temperatures which bound the peak cladding temperatures. We conservatively assume that the peak clad temperature is equal to the peak pellet centerline temperature.

The following additional observations can be derived by inspecting the temperature field obtained from the finite volume analysis:

- The maximum fuel cladding temperature is well within the ISG-11 [4.3.8] recommended temperature limits.

- The maximum temperature of the basket structural material is within the stipulated Design Temperature.

- The maximum temperature of the neutron absorber is below the stipulated limit.

- The maximum temperatures of the MPC pressure boundary materials are well below their respective ASME Code limits.

- The maximum temperatures of the overpack pressure boundary material are well below their respective ASME Code limits.

- The neutron shielding material (Holtite-A) will not experience temperatures in excess of its qualified limit.

- The local temperatures of the mechanical seals are well below their respective long-term limits (Table 4.3.1).

The above observations lead us to conclude that the temperature field in the HI-STAR 100 System with a fully loaded MPC containing design-basis heat emitting SNF complies with all regulatory and industry temperature limits. In other words, the thermal environment in the HI-STAR 100 System will be conducive to long-term safe storage of spent nuclear fuel.

4.4.2.2 Maximum MPC Basket Temperature Under Drying Conditions

A plot of typical steady-state temperature contours under vacuum drying conditions for MBF is shown in Figure 4.4.19. The peak fuel clad temperature during short-term vacuum drying operations for MBF will be limited to less than the applicable design limit for all baskets at design basis maximum heat loads.

HBF must be dried using an FHD which, as described in Appendix 4.A, will actively maintain the peak fuel clad temperature to less than the applicable design limit for all baskets at design basis maximum heat loads.

4.4.3 Minimum Temperatures

In Table 2.2.2 of this report, the minimum ambient temperature condition required to be considered for HI-STAR 100 System design is specified to be -40°F. If, conservatively, a zero decay heat load (with no solar input) is applied to the stored fuel assemblies then every component of the system at steady state would be at this minimum ambient temperature. All HI-STAR 100 System materials of construction would satisfactorily perform their intended function in the storage mode at this minimum postulated temperature condition. Structural evaluations in Chapter 3 show the acceptable performance of the overpack and MPC steel material at low temperature. Criticality and shielding functions of the HI-STAR 100 System materials (Chapters 5 and 6) are unaffected by exposure to this minimum temperature.

4.4.4 Maximum Internal Pressure

The MPC is initially filled with helium after fuel loading and drying prior to installing the MPC closure ring. During normal storage, the gas temperature within the MPC rises to its maximum operating basis temperature as determined based on the thermal analysis methodology described earlier. The gas pressure inside the MPC will also increase with rising temperature. The pressure rise is determined using the ideal gas law which states that the absolute pressure of a fixed volume of confined gas is proportional to its absolute temperature. In Tables 4.4.13 and 4.4.14, a summary of calculations for determining the net free volume in the MPC-24 and MPC-68 are presented. Analogous calculations for the MPC-32 are presented in Table 3.4.13 of the HI-STAR 100 transport SAR.

The maximum gas pressure in the MPC is considered for a postulated accidental release of fission product gases caused by fuel rods rupture. For these fuel rod rupture conditions, the amounts of each of the release gas constituents in the MPC cavity are summed and the resulting total pressures determined from the Ideal Gas Law. Based on fission gases release fractions (per NUREG-1536 criteria [4.1.3]), minimum net free volume and maximum initial fill gas pressure, bounding maximum gas pressures with 1% (normal), 10% (off-normal), and 100% (accident condition) rod rupture are given in Table 4.4.15 for vertically-oriented casks and in Table 3.4.15 of the HI-STAR 100 transport SAR for horizontally-oriented casks. The MPC maximum gas pressures listed in Table 4.4.15 and in Table 3.4.15 of the HI-STAR 100 transport SAR are all below the MPC design internal pressure listed in Table 2.2.1.

The inclusion of PWR non-fuel hardware (BPRA control elements and thimble plugs) to the MPC-24 and MPC-32 influences the internal pressure in two ways. The presence of non-fuel hardware enhances heat dissipation, thus lowering fuel temperatures and the gas filling the space between fuel rods. The gas volume displaced by the mass of non-fuel hardware lowers the cavity free volume. These two effects, namely, temperature lowering and free volume reduction, have opposing influence in the MPC cavity pressure. The first effect lowers gas pressure while the second effect raises it. In the HI-STAR thermal analysis, the computed temperature field (with non-fuel hardware excluded) provides conservatively bounding temperature fields. The MPC cavity free space is computed based on volume displacement by the heaviest fuel (bounding weight) with non-fuel hardware included).

During in-core irradiation of BPRAs, the B-10 isotope in the neutron absorbing material is transformed to helium atoms. Two different forms of the neutron absorbing material are used in BPRAs: Borasilicate glass and B_4C in a refractory solid matrix (Al_2O_3). Borosilicate glass (primarily a constituent of Westinghouse BPRAs) is used in the shape of hollow pyrex glass tubes sealed within steel rods and supported on the inside by a thin walled steel liner. To accommodate helium diffusion from the glass rod into the rod internal space, a relatively high void volume (~40%) is engineered in this type of rod design. The rod internal pressure is thus designed to remain below reactor operating conditions (2,300 psia and approximately 600ºF coolant temperature). The B_4C- Al_2O_3 neutron absorber material is principally used in B&W and CE fuel BPRA designs. The relatively low temperature of the poison material in BPRA rods (relative to fuel pellets) favor the entrapment of helium atoms in the solid matrix.

Several BPRA designs are used in PWR fuel which differ in the number, diameter, and length of poison rods. The older Westinghouse fuel (W-14x14 and AW-15x15) has used 6, 12, 16, and 20 rods per assembly BPRAs and the later (W-17x17) fuel uses up to 24 rods per BPRA. The BPRA rods in the older fuel are much larger than the later fuel and, therefore, the B-10 isotope inventory in the 20-rod BPRAs bound the newer W-17x17 fuel. Based on bounding BPRA rods internal pressure, a large hypothetical quantity of helium (7.2 g-moles/BPRA) is assumed to be available for release into the MPC cavity from each PWR fuel assembly. To accommodate this quantity of helium gas[†] at the NUREG-1536 stipulated rods rupture assumptions, the initial helium backfill is reduced such that the final confinement boundary pressures are approximately unchanged from inclusion of non-fuel hardware. The MPC cavity pressures are summarized in Table 4.4.15

4.4.5 Maximum Thermal Stresses

Thermal expansion induced mechanical stresses due to the non-uniform temperature distribution are reported in Chapter 3. Table 4.4.16 provides a summary of HI-STAR 100 System component temperature inputs for structural evaluation.

Table 4.4.22 provides a summary of confinement boundary temperatures during normal storage conditions for vertically-oriented casks. Analogous temperatures for the helium-retention boundary components in horizontally-oriented casks are presented in Table 3.4.24 of the HI-STAR 100 transport SAR. Structural evaluation in Section 3.4.4 references these temperature results to demonstrate confinement boundary integrity.

4.4.6 Evaluation of System Performance for Normal Conditions of Storage

The HI-STAR 100 System thermal analysis is based on a detailed and complete heat transfer model which properly accounts for radiation, conduction and natural convection modes of heat

†

transfer in various portions of the MPC and overpack. The thermal model incorporates many conservative features that are listed below:

1. The most severe levels of environmental factors - bounding long-term annual ambient temperature with insolation - were coincidentally imposed on the HI-STAR 100 cask. A bounding solar absorbtivity of 1.0 was applied to all surfaces exposed to insolation.

2. No credit was considered for the thermosiphon heat transfer which is intrinsic to the HI-STAR fuel baskets.

3. The most adversely located HI-STAR 100 System in an ISFSI array was considered for analysis.

4. No credit was considered for conduction through the radial neutron shielding material.

5. For vertically-oriented casks, a uniform nominal radial gap between overpack-to-MPC was applied to the cask thermal model. No credit for gap reduction due to differential thermal expansion under the hot condition was considered. The MPC is considered to be in concentric alignment inside the overpack cavity. This is a worst case scenario since any eccentricity will improve conductive heat transport in this region.

6. Not Used.

7. Interfacial contact conductance of multilayered intermediate shell contacting layers was conservatively determined to bound surface finish, contact pressure, and base metal conductivity conditions.

8. No credit was considered for contact between fuel assemblies and the MPC basket wall or between the MPC basket and the basket supports. The fuel assemblies and MPC basket were conservatively considered to be in concentric alignment.

9. The MPC is assumed to be loaded with the SNF type which has the maximum equivalent thermal resistance of all fuel types in its category (BWR or PWR), as applicable.

10. The decay heat load, which is a function of burnup and decay time, varies in a narrow range within the group of PWR assemblies considered (Table 4.4.5) and also within the group of BWR assemblies considered (Table 4.4.6). The assembly type which gives the maximum decay heat load for a given burnup is used for defining the decay heat load vs. decay time. The B&W 15×15 is the limiting PWR SNF type (see Table 2.1.5). The governing BWR fuel is GE 7×7 (see Table 2.1.5). For other than the governing fuel types, there is a small conservatism in the decay heat load term.

11. The MPC basket axial conductivity is conservatively assumed to be equal to the lower basket cross sectional effective conductivity.

12. Deleted.

Temperature distribution results obtained from the conservative thermal models show that the established maximum fuel cladding temperature limits are met with adequate margins. Expected margins during normal storage will be larger due to the many conservative assumptions incorporated in the analysis. The long-term impact of decay heat induced temperature levels on the HI-STAR 100 System structural and neutron shielding materials is considered to be negligible. The maximum local MPC basket temperature level is below the recommended limits for structural materials in terms of susceptibility to stress, corrosion and creep induced degradation. Furthermore, structural evaluation (Chapter 3) has demonstrated that stresses (including those induced due to imposed temperature gradients) are within ASME B&PV Code limits. The maximum local neutron shield temperature is lower than design limits. Section 4.5 provides a discussion of compliance with regulatory criteria 1 through 8 listed in Section 4.0. The above-mentioned considerations lead to the conclusion that the HI-STAR 100 System thermal design is in compliance with 10CFR72 requirements.

Table 4.4.1

CLOSED CAVITY NUSSELT NUMBER RESULTS
FOR HELIUM-FILLED MPC PERIPHERAL VOIDS
IN VERTICALLY-ORIENTED SYSTEMS

Temperature [ºF / ºC]	Nusselt Number (PWR Baskets)	Nusselt Number (BWR Basket)
200 / 93	3.17	2.41
450 / 232	2.56	1.95
700 / 371	2.21	1.68

Table 4.4.2

RELATIONSHIP BETWEEN HI-STAR 100 SYSTEM REGIONS
AND MATHEMATICAL MODEL DESCRIPTIONS

HI-STAR System Region	Mathematical Model	Subsections
Fuel Assembly	Fuel Region Effective Thermal Conductivity	4.4.1.1.2
MPC	Effective Thermal Conductivity of Boral/Sheathing/Box Wall Sandwich	4.4.1.1.3
	Basket In-Plane Conductive Heat Transport	4.4.1.1.4
	Heat Transfer in MPC Basket Peripheral Region	4.4.1.1.5
	Effective Thermal Conductivity of Flexible MPC Basket-to-Shell Aluminum Heat Conduction Elements	4.4.1.1.10
Overpack	Effective Conductivity of Multilayered Intermediate Shell Region	4.4.1.1.6
	Effective Thermal Conductivity of Holtite Neutron Shielding Region	4.4.1.1.9
Ambient Environment	Heat Rejection from Overpack Exterior Surfaces	4.4.1.1.7
	Solar Heat Input	4.4.1.1.8
Assembled Cask Model	Overview of the Thermal Model	4.4.1.1.1
	FLUENT Model for HI-STAR 100	4.4.1.1.11

Table 4.4.3

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Table 4.4.4

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Table 4.4.5

SUMMARY OF PWR FUEL ASSEMBLY EFFECTIVE THERMAL CONDUCTIVITIES

	Fuel	Btu/ ft-hr-ºF [W/m-K]		
		@ 200ºF (93ºC)	@ 450ºF (272 ºC)	@ 700ºF (371 ºC)
1	W - 17×17 OFA	0.182 [0.315]	0.277 [0.479]	**0.402 [0.696]**
2	W - 17×17 Std	0.189 [0.327]	0.286 [0.495]	0.413 [0.715]
3	W - 17×17 Vantage	0.182 [0.315]	0.277 [0.479]	0.402 [0.696]
4	W - 15×15 Std	0.191 [0.331]	0.294 [0.509]	0.430 [0.744]
5	W - 14×14 Std	0.182 [0.315]	0.284 [0.492]	0.424 [0.734]
6	W - 14×14 OFA	**0.175 [0.303]**	**0.275 [0.476]**	0.413 [0.715]
7	B&W - 17×17	0.191 [0.331]	0.289 [0.500]	0.416 [0.720]
8	B&W - 15×15	0.195 [0.338]	0.298 [0.516]	0.436 [0.755]
9	CE - 16×16	0.183 [0.317]	0.281 [0.486]	0.411 [0.711]
10	CE - 14×14	0.189 [0.327]	0.293 [0.507]	0.435 [0.753]
11	HN† -15×15 SS	0.180 [0.312]	0.265 [0.459]	0.370 [0.640]
12	W-14×14 SS	0.170 [0.294]	0.254 [0.440]	0.361 [0.625]
13	B&W – 15x15	0.187 [0.324]	0.289 [0.500]	0.424 [0.734]
14	CE-14x14 (MP2)	0.188 [0.325]	0.293 [0.507]	0.434 [0.751]

Note: Boldface values denote the lowest thermal conductivity in each column.

† Haddam Neck B&W or Westinghouse stainless steel clad fuel assemblies.

Table 4.4.6
SUMMARY OF BWR FUEL ASSEMBLY EFFECTIVE
THERMAL CONDUCTIVITIES

	Fuel	Btu/ft-hr-ºF [W/m-K]		
		@ 200ºF (93ºC)	@ 450ºF (272ºC)	@ 700ºF (371ºC)
1	Dresden 1 - 8×8†	0.119 [0.206]	0.201 [0.348]	0.319 [0.552]
2	Dresden 1 - 6×6†	0.126 [0.218]	0.215 [0.372]	0.345 [0.597]
3	GE - 7×7	0.171 [0.296]	0.286 [0.495]	0.449 [0.777]
4	GE - 7×7R	0.171 [0.296]	0.286 [0.481]	0.449 [0.777]
5	GE - 8×8	0.168 [0.291]	0.278 [0.476]	0.433 [0.750]
6	GE - 8×8R	**0.166 [0.287]**	0.275 [0.476]	0.430 [0.744]
7	GE10 - 8×8	0.168 [0.291]	0.280 [0.485]	0.437 [0.756]
8	GE11 - 9×9	0.167 [0.289]	**0.273 [0.473]**	**0.422 [0.730]**
9	AC†† -10×10 SS	0.152 [0.263]	0.222 [0.384]	0.309 [0.535]
10	Exxon-10×10 SS	0.151 [0.261]	0.221 [0.383]	0.308 [0.533]
11	Humboldt Bay-7x7†	0.127 [0.220]	0.215 [0.372]	0.343 [0.594]
12	Dresden-1 Thin †Clad-6x6	0.124 [0.215]	0.212 [0.367]	0.343 [0.594]
13	Damaged Dresden-1 8×8† (in a damaged fuel container)	0.107 [0.185]	0.169 [0.293]	0.254 [0.440]
14	Damaged† Dresden-1 8x8 (in TN D-1 canister)	0.107 [0.185]	0.168 [0.291]	0.252 [0.436]
15	8x8 QUAD+ Westinghouse†	0.164 [0.284]	0.276 [0.478]	0.435 [0.753]

Note: Boldface values denote the lowest thermal conductivity in each column.

† Fuel cladding temperatures for low heat emitting (intact and damaged) fuel types in the HI-STAR 100 System will be bounded by design basis fuel cladding temperatures. Therefore, these fuel assembly types are excluded from the list of design basis fuel assemblies (Zircaloy clad) evaluated to determine the most resistive SNF type.

†† Allis-Chalmers stainless steel clad fuel assemblies.

Table 4.4.7

MPC BASKET EFFECTIVE THERMAL CONDUCTIVITY VALUES†

Basket	Btu/ft-hr-ºF [W/m-K]		
	@200ºF (93ºC)	@450ºF (232ºC)	@700ºF (371ºC)
MPC-24 (Zircaloy Clad Fuel)	1.108 [1.92]	1.495 [2.59]	1.954 [3.38]
MPC-68 (Zircaloy Clad Fuel)	0.959 [1.66]	1.188 [2.06]	1.432 [2.48]
MPC-24 (Stainless Steel Clad Fuel)	0.995 [1.72]	1.321 [2.29]	1.700 (a) [2.94]
MPC-68 (Stainless Steel Clad Fuel)	0.931 [1.61]	1.125 [1.95]	1.311 (b) [2.27]
MPC-68 (Dresden-1 8x8 in canister)	0.861 [1.49]	1.055 [1.83]	1.242 [2.15]

(a) Conductivity is 13% less than corresponding Zircaloy fueled basket.
(b) Conductivity is 9% less than corresponding Zircaloy fueled basket.

† The values reported in this table are conservatively understated.

Table 4.4.8

EFFECTIVE THERMAL CONDUCTIVITY OF THE NEUTRON SHIELD/RADIAL CHANNEL LEG REGION

Condition/Temperature (ºF [ºC])	Thermal Conductivity (Btu/ft-hr-ºF [W/m-K])
Normal condition:	
200 [93]	1.953 [3.38]
450 [232]	1.812 [3.14]
700 [371]	1.645 [2.85]
Fire condition:	
200 [93]	3.012 [5.21]
450 [232]	2.865 [4.96]
700 [371]	2.689 [4.65]

Table 4.4.9

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Table 4.4.10

HI-STAR 100 SYSTEM LONG-TERM VERTICAL ORIENTATION
NORMAL STORAGE† MAXIMUM TEMPERATURES
(24-PWR ASSEMBLIES, MPC)

	Maximum Temperature (°F [°C])	Normal Condition Design Temperature (°F [°C])
Fuel Cladding	709 [376]	752 [400]
MPC Basket Centerline	675 [357]	725 [385]
MPC Basket Periphery	451 [233]	725 [385]
MPC Outer Shell Surface	332 [167]	350 [177]
MPC/Overpack Helium Gap Outer Surface	292 [144]	400 [204]
Neutron Shield Inner Surface	274 [134]	300 [149]
Overpack Outer Enclosure Surface	229 [109]	350 [177]
Overpack Bolted Closure Plate††	155 [68]	400 [204]
Overpack Bottom Plate††	241 [116]	350 [177]

† Ambient Temperature = 80ºF (27°C)
Cask Array Pitch = 3 x Cask Radius = 12 ft. (3.66 m)

†† Overpack closure plate and vent/drain port plug seals normal condition design temperature is 400°F (204°C). The maximum seals temperatures are bounded by the reported closure plate and bottom plate maximum temperatures. Consequently, a large margin of safety exists to permit safe operation of seals in the overpack helium retention boundary.

Table 4.4.11

HI-STAR 100 SYSTEM LONG-TERM VERTICAL ORIENTATION
NORMAL STORAGE† MAXIMUM TEMPERATURES
(68-BWR ASSEMBLIES, MPC)

	Maximum Temperature (°F [°C])	Normal Condition Design Temperature (°F [°C])
Fuel Cladding	741 [394]	752 [400]
MPC Basket Centerline	725 [385]	725 [385]
MPC Basket Periphery	393 [201]	725 [385]
MPC Outer Shell Surface	331 [166]	350 [177]
MPC/Overpack Helium Gap Outer Surface	292 [144]	400 [204]
Neutron Shield Inner Surface	273 [134]	300 [149]
Overpack Outer Enclosure Surface	228 [109]	350 [177]
Overpack Bolted Closure Plate††	155 [68]	400 [204]
Overpack Bottom Plate††	213 [101]	350 [177]

† Ambient Temperature = 80ºF (27°C)
Cask Array Pitch = 3 x Cask Radius = 12 ft. (3.66 m)

†† Overpack closure plate and vent/drain port plug seals normal condition design temperature is 400°F (204°C). The maximum seals temperatures are bounded by the reported closure plate and bottom plate maximum temperatures. Consequently, a large margin of safety exists to permit safe operation of seals in the overpack helium retention boundary.

Table 4.4.12

HI-STAR 100 SYSTEM LONG-TERM VERTICAL ORIENTATION
NORMAL STORAGE† MAXIMUM TEMPERATURES
(32-PWR ASSEMBLIES, MPC)

	Maximum Temperature (°F [°C])	Normal Condition Design Temperature (°F [°C])
Fuel Cladding	710 [377]	752 [400]
MPC Basket Centerline	677 [358]	725 [385]
MPC Basket Periphery	382 [194]	725 [385]
MPC Outer Shell Surface	315 [157]	350 [177]
MPC/Overpack Helium Gap Outer Surface	275 [135]	400 [204]
Neutron Shield Inner Surface	251 [122]	300 [149]
Overpack Outer Enclosure Surface	212 [100]	350 [177]
Overpack Bolted Closure Plate††	149 [65]	400 [204]
Overpack Bottom Plate††	219 [104]	350 [177]

† Ambient Temperature = 80ºF (27°C)
Cask Array Pitch = 3 x Cask Radius = 12 ft. (3.66 m)

†† Overpack closure plate and vent/drain port plug seals normal condition design temperature is 400°F (204°C). The maximum seals temperatures are bounded by the reported closure plate and bottom plate maximum temperatures. Consequently, a large margin of safety exists to permit safe operation of seals in the overpack helium retention boundary.

Table 4.4.13

SUMMARY OF MPC-24 FREE VOLUME CALCULATIONS

Item	**Volume (ft^3 [L])**
Cavity Volume	367.9 [10418]
Basket Metal Volume	37.9 [1073]
Bounding Fuel Assemblies Volume	78.8 [2231]
Basket Supports and Fuel Spacers Volume	6.1 [173]
Aluminum Conduction Elements	5.9† [167]
Net Free Volume	237.5 [6,725]

† Bounding 1,000 lbs (455 kg) weight.

Table 4.4.14

SUMMARY OF MPC-68 FREE VOLUME CALCULATIONS

Item	Volume (ft^3 [L])
Cavity Volume	367.3 [10401]
Basket Metal Volume	34.8 [985]
Bounding Fuel Assemblies Volume	93.0 [2633]
Basket Supports and Fuel Spacers Volume	11.3 [320]
Aluminum Conduction Elements	5.9† [167]
Net Free Volume	222.3 [6,295]

† Bounding 1,000 lbs (455 kg) weight.

Table 4.4.15

SUMMARY OF MPC CONFINEMENT BOUNDARY PRESSURES† FOR NORMAL VERTICAL ORIENTATION LONG-TERM STORAGE

Condition	Pressure (psig [kPa])
MPC-24††:	
Initial backfill (at 70ºF / 21ºC)	22.2 [153]
Normal condition	43.8 [302]
With 1% rods rupture	44.3 [305]
With 10% rods rupture	49.1 [339]
With 100% rods rupture	97.3 [671]
MPC-68:	
Initial backfill (at 70ºF / 21ºC)	28.5 [197]
Normal condition	57.5 [396]
With 1% rods rupture	57.8 [399]
With 10% rods rupture	60.2 [415]
With 100% rods rupture	84.6 [583]
MPC-32††:	
Initial backfill (at 70ºF / 21ºC)	20.3 [140]
Normal condition	42.9 [296]
With 1% rods rupture	43.4 [299]
With 10% rods rupture	48.1 [332]
With 100% rods rupture	99.0 [683]

† Pressure analysis is based on NUREG-1536 criteria (i.e., 100% of rods fill gas and 30% of radioactive gases are available for release from a ruptured rod).

†† PWR fuel storage includes hypothetical BPRA rods rupture in the pressure calculations.

†† PWR fuel storage includes hypothetical BPRA rods rupture in the pressure calculations.

Table 4.4.16

SUMMARY OF HI-STAR 100 SYSTEM COMPONENTS
NORMAL VERTICAL ORIENTATION STORAGE TEMPERATURES (ºF [ºC])]

Location	**MPC-24**	**MPC-68**
MPC Basket Top:		
Basket center	180 [82]	179 [82]
Basket periphery	168 [76]	168 [76]
MPC shell	166 [74]	167 [75]
Overpack inner shell	162 [72]	163 [73]
Overpack enclosure shell	159 [71]	160 [71]
MPC Basket Bottom:		
Basket center	251 [122]	220 [104]
Basket periphery	226 [108]	204 [96]
MPC shell	222 [106]	203 [95]
Overpack inner shell	218 [103]	201 [94]
Overpack enclosure shell	177 [81]	167 [75]

Table 4.4.17

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Table 4.4.18

VERTICAL ORIENTATION MPC-24 BASKET PEAK FUEL CLADDING TEMPERATURE AS A FUNCTION OF TOTAL HEAT LOAD

Total Basket Decay Heat Load (kW)	Peak Cladding Temperature (°F [°C])
19.0†	708.8 [376]
18.5	696.9 [369]
17.0	660.1 [349]
15.5	621.9 [328]

† Design Basis Maximum (equivalent to 792 watts per assembly).

Table 4.4.19

VERTICAL ORIENTATION MPC-68 BASKET PEAK CLADDING TEMPERATURE AS A FUNCTION OF TOTAL DECAY HEAT LOAD

Total Basket Decay Heat Load (kW)	Peak Cladding Temperature (ºF [ºC])
18.5†	741.5 [394]
17.5	713.6 [379]
15.5	656.2 [347]

† Design Basis Maximum (equivalent to 272 watts per assembly).

Table 4.4.20

SUMMARY OF LOADED HI-STAR SYSTEM
BOUNDING COMPONENT WEIGHTS AND THERMAL INERTIAS

Component	Weight (lbs [kg])	Heat Capacity (Btu/lb-°F [J/kg-K])	Thermal Inertia (Btu/°F [kJ/K])
Holtite-A	11,000 [5,000]	0.39 [1,633]	4,290 [8,147]
Carbon Steel	140,000 [63,636]	0.1 [419]	14,000 [29,586]
Alloy-X MPC (empty)	35,000 [15,909]	0.12 [502]	4,200 [7,976]
Fuel	40,000 [18,182]	0.056 [234]	2,240 [4,254]
MPC Cavity Water†	6,500 [2,955]	1.0 [4,187]	6,500 [12,344]
			31,230 [59,306] (Total)

† Based on smallest MPC-68 cavity net free volume with 50% credit for flooded water mass.

Table 4.4.21

MAXIMUM ALLOWABLE TIME DURATION FOR WET TRANSFER OPERATIONS

Initial Temperature (ºF [ºC])	Time Duration (hr)
115 [46]	44.2
120 [49]	42.0
125 [52]	39.7
130 [54]	37.4
135 [57]	35.1
140 [60]	32.8
145 [63]	30.5
150 [66]	28.3

Table 4.4.22

SUMMARY OF MPC CONFINEMENT BOUNDARY TEMPERATURE DISTRIBUTION DURING NORMAL VERTICAL ORIENTATION STORAGE CONDITIONS

Location	Figure 3.4.44 Designation	MPC-24 (ºF [ºC])	MPC-68 (ºF ºC])
MPC Lid Inside Surface at Centerline	A	179 [82]	178 [81]
MPC Lid Outside Surface at Centerline	B	173 [78]	172 [78]
MPC Lid Inside Surface at Periphery	C	166 [74]	167 [75]
MPC Lid Outside Surface at Periphery	D	164 [73]	164 [73]
MPC Baseplate Inside Surface at Centerline	E	249 [121]	218 [103]
MPC Baseplate Outside Surface at Centerline	F	241 [116]	213 [101]
MPC Baseplate Inside Surface at Periphery	G	222 [106]	203 [95]
MPC Baseplate Outside Surface at Periphery	H	219 [104]	200 [93]
MPC Shell Maximum	I	332 [167]	331 [166]

Table 4.4.23

SUMMARY OF 10×10 ARRAY TYPE BWR FUEL ASSEMBLY EFFECTIVE THERMAL CONDUCTIVITIES†

FUEL	Btu/ft-hr-ºF [W/m-K]		
	@200ºF (93ºC)	@450ºF (232ºC)	@700ºF (371ºC)
GE-12/14	0.166 [0.287]	0.269 [0.466]	0.412 [0.713]
Atrium-10	0.164 [0.284]	0.266 [0.460]	0.409 [0.708]
SVEA-96	0.164 [0.284]	0.269 [0.466]	0.416 [0.720]

† The conductivities reported in this table are obtained by the simplified method described in the beginning of the Subparagraph 4.4.1.1.2.

Table 4.4.24

COMPARISON OF ATRIUM-10 BWR FUEL ASSEMBLY CONDUCTIVITY† WITH THE BOUNDING†† BWR FUEL ASSEMBLY CONDUCTIVITY

Temperature (°F [°C]	Atrium-10 BWR Assembly (Btu/ft-hr-°F [W/m-K])	Bounding BWR Assembly (Btu/ft-hr-°F [W/m-K])
200 [93]	0.225 [0.389]	0.171 [0.296]
450 [232]	0.345 [0.597]	0.271 [0.469]
700 [371]	0.504 [0.872]	0.410 [0.710]

† The reported effective conductivity has been obtained from a rigorous finite element model.

†† The bounding BWR fuel assembly conductivity applied in the MPC-68 basket thermal analysis.

Table 4.4.25

PLANT SPECIFIC BWR FUEL TYPES EFFECTIVE THERMAL CONDUCTIVITY*

Fuel	Btu/ft-hr-ºF [W-m-K]		
	@200ºF (93ºC)	@ 450ºF (232ºC)	@ 700ºF (371ºC)
Oyster Creek (7x7)	0.165 [0.286]	0.273 [0.472]	0.427 [0.739]
Oyster Creek (8x8)	0.162 [0.280]	0.266 [0.460]	0.413 [0.715]
TVA Browns Ferry (8x8)	0.160 [0.277]	0.264 [0.457]	0.411 [0.711]
SPC-5 (9x9)	0.149 [0.258]	0.245 [0.424]	0.380 [0.658]

* The conductivities reported in this table are obtained by a simplified analytical method described in Paragraph 4.4.1.2.

Table 4.4.26

COMPARISON OF TRANSPORTATION RESULTS INCORPORATED BY REFERENCE WITH NORMAL HORIZONTAL STORAGE ON SUPPORT STRUCTURE

Normal Transport Incorporated by Reference vs. Normal Storage and Allowables HI-STAR 100 with BWR Canister			
	Normal Horizontal Transport Note 1	**Normal Horizontal Storage** Note 2	**Normal Storage Allowable** Note 3
Ambient Temperature			
Normal Condition	100°F / 38°C	80°F / 27°C	N/A
Component Temperatures			
Fuel Cladding	713°F / 378°C	639°F / 337°C	752°F / 400°C
MPC Shell	306°F / 152°C	288°F / 142°C	450°F / 232°C
Internal Pressure Note 4			
MPC Cavity Pressure	85.8 psig / 592 kPa	85 psig / 586 kPa	100 psig / 689 kPa
HI-STAR 100 with PWR Canister			
	Normal Horizontal Transport Note 5	**Normal Horizontal Storage** Note 2	**Normal Storage Allowable** Note 3
Ambient Temperature			
Normal Condition	100°F / 38°C	80°F / 27°C	N/A
Component Temperatures			
Fuel Cladding	701°F / 372°C	698°F / 370°C	752°F / 400°C
MPC Shell	315°F / 157°C	306°F / 152°C	450°F / 232°C
Internal Pressure Note 4			
MPC Cavity Pressure	89.3 psig / 616 kPa	89 psig / 614 kPa	100 psig / 689 kPa

Notes

1. The temperature and MPC cavity pressure results for normal transport are incorporated by reference from Tables 3.4.11 and 3.4.15 of HI-STAR 100 SAR [2].
2. Thermal evaluations of a horizontal HI-STAR 100 cask positioned on the saddle supports was performed under the Licensing Basis heat loads. These evaluations are performed using the methodology described in Section 3.4 of HI-STAR 100 SAR [2].
3. The temperature and pressure limits are extracted from Tables 2.2.3 and 2.2.1 of HI-STAR 100 FSAR [1].
4. The initial helium backfill pressure used to compute the confinement boundary pressures is 42.8 psia (295 kPa) at a 70°F (21°C) reference temperature, the maximum backfill level from Table 2-1 in Appendix A of the proposed technical specifications.
5. The temperature and MPC cavity pressure results for normal transport are incorporated by reference from Tables 3.4.10 and 3.4.15 of HI-STAR 100 SAR [2].

TABLE 4.4.27

COMPARISON OF TRANSPORTATION RESULTS INCORPORATED BY REFERENCE WITH NORMAL HORIZONTAL STORAGE OF AN ARRAY OF CASKS

Normal Horizontal Storage In An Array HI-STAR 100 with BWR Canister		
	Normal Horizontal Storage (Note 1)	**Normal Storage Allowable (Note 2)**
Component Temperatures		
Fuel Cladding	644°F / 340°C	752°F / 400°C
MPC Shell	298°F / 148°C	450°F / 232°C
MPC Baseplate	281°F / 138°C	400°F / 204°C
MPC Lid	180°F / 82°C	550°F / 288°C
Internal Pressure Note 3		
MPC Cavity Pressure	86 psig / 593 kPa	100 psig / 689 kPa
HI-STAR 100 with PWR Canister		
	Normal Horizontal Storage (Note 1)	**Normal Storage Allowable (Note 2)**
Component Temperatures		
Fuel Cladding	703°F / 373°C	752°F / 400°C
MPC Shell	316°F / 158°C	450°F / 232°C
MPC Baseplate	300°F / 149°C	400°F / 204°C
MPC Lid	180°F / 82°C	550°F / 288°C
Internal Pressure Note3		
MPC Cavity Pressure	90 psig / 621 kPa	100 psig / 689 kPa

Notes

1. Thermal evaluations of a horizontal HI-STAR 100 cask positioned on the saddle supports was performed under the Licensing Basis heat loads. These evaluations are performed using the methodology described in Section 3.4 of HI-STAR 100 SAR [2].
2. The temperature and pressure limits are extracted from Tables 2.2.3 and 2.2.1 of HI-STAR 100 FSAR [1].
3. The initial helium backfill pressure used to compute the confinement boundary pressures is 42.8 psia (295 kPa) at a 70°F (21°C) reference temperature, the maximum backfill level from Table 2-1 in Appendix A of the proposed technical specifications.

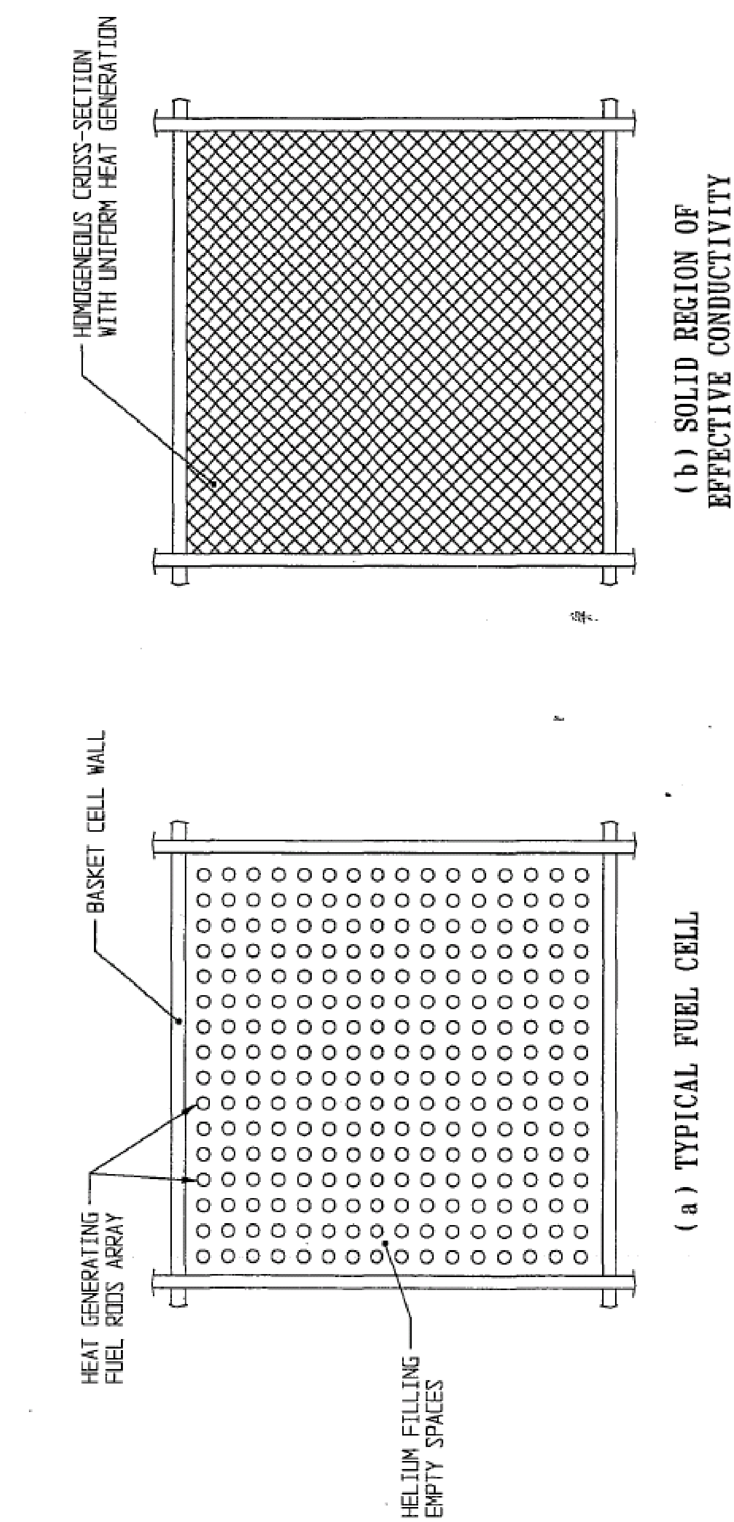

FIGURE 4.4.1; HOMOGENIZATION OF THE STORAGE CELL CROSS-SECTION

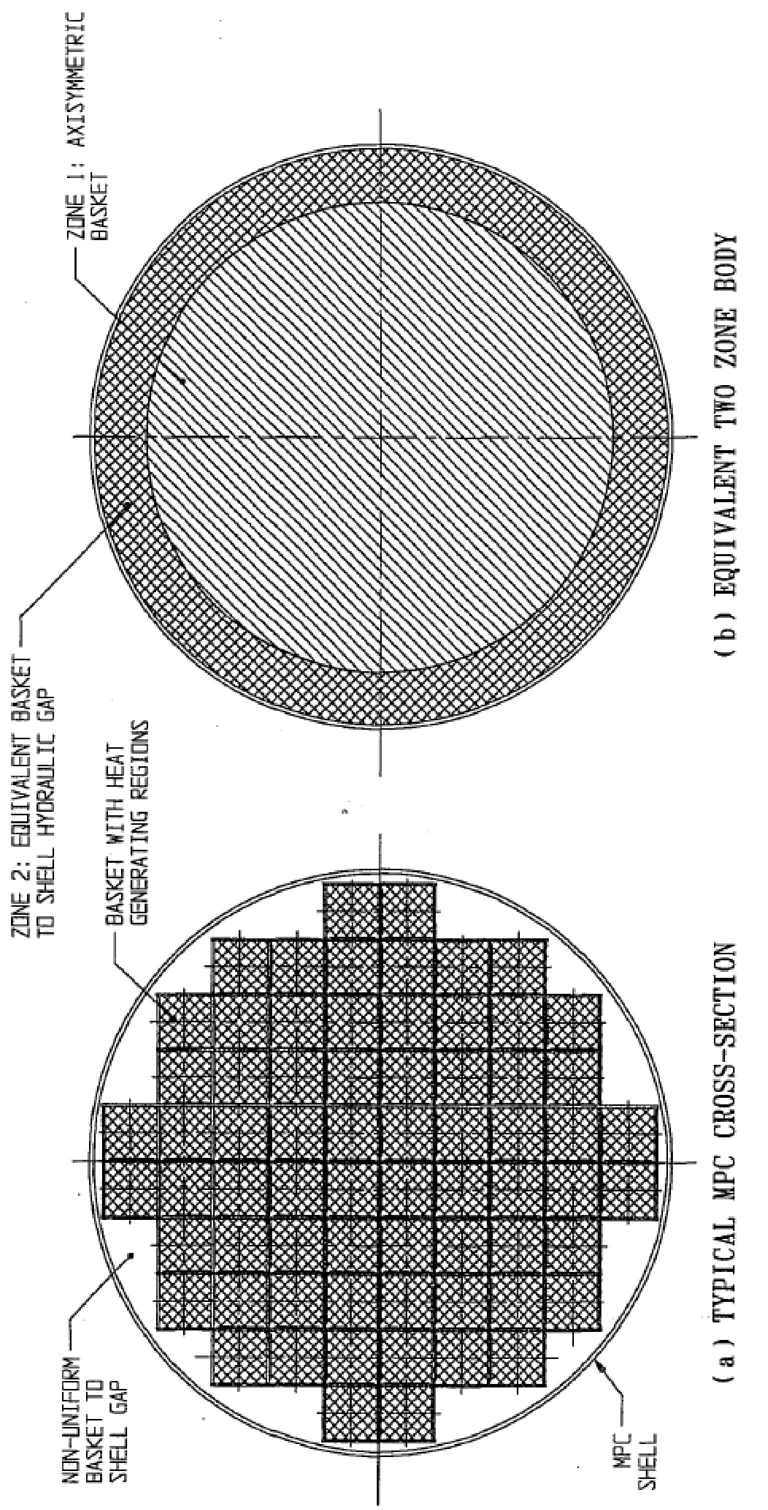

FIGURE 4.4.2; MPC CROSS-SECTION REPLACED WITH AN EQUIVALENT TWO ZONE AXISYMMETRIC BODY

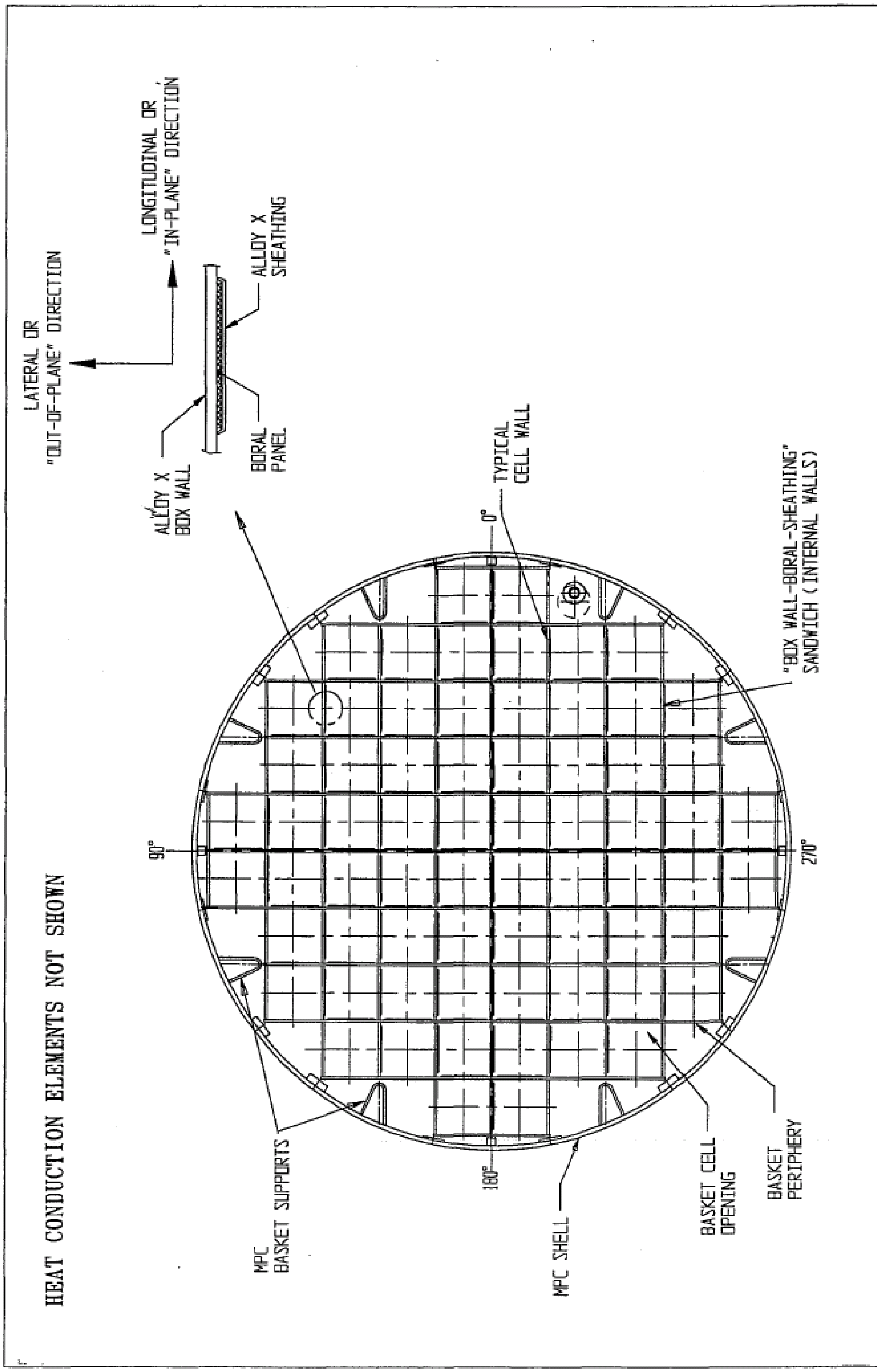

FIGURE 4.4.3; TYPICAL MPC BASKET PARTS IN A CROSS-SECTIONAL VIEW

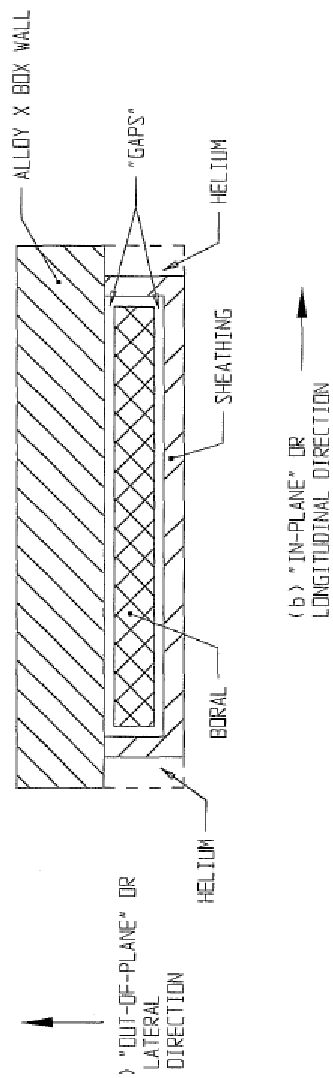

FIGURE 4.4.4; "BOX WALL-BORAL-SHEATHING" SANDWICH

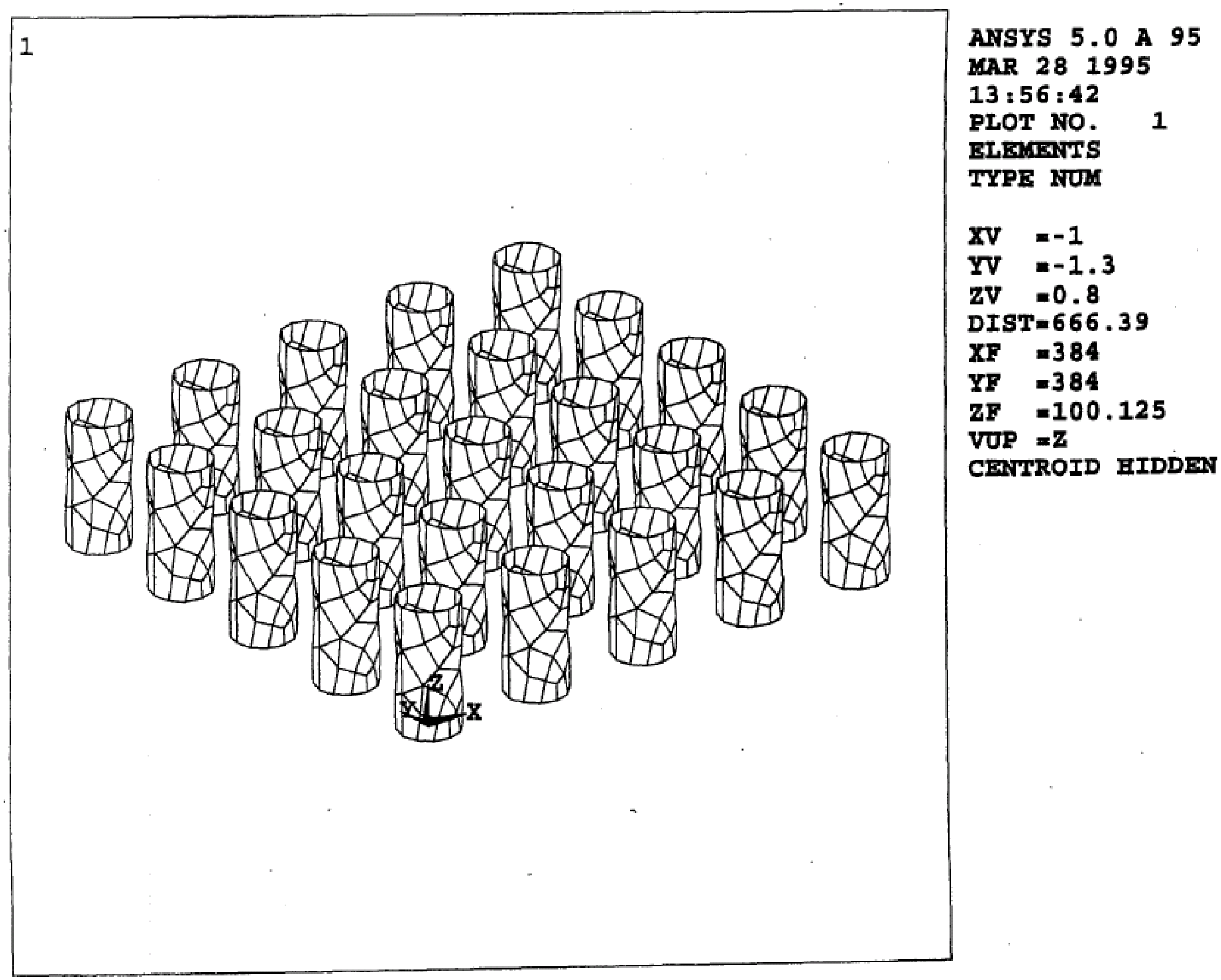

FIGURE 4.4.5; ANSYS FINITE ELEMENT MODEL FOR EVALUATION OF RADIATIVE BLOCKING FACTOR FOR A CASK ARRAY AT AN ISFSI SITE

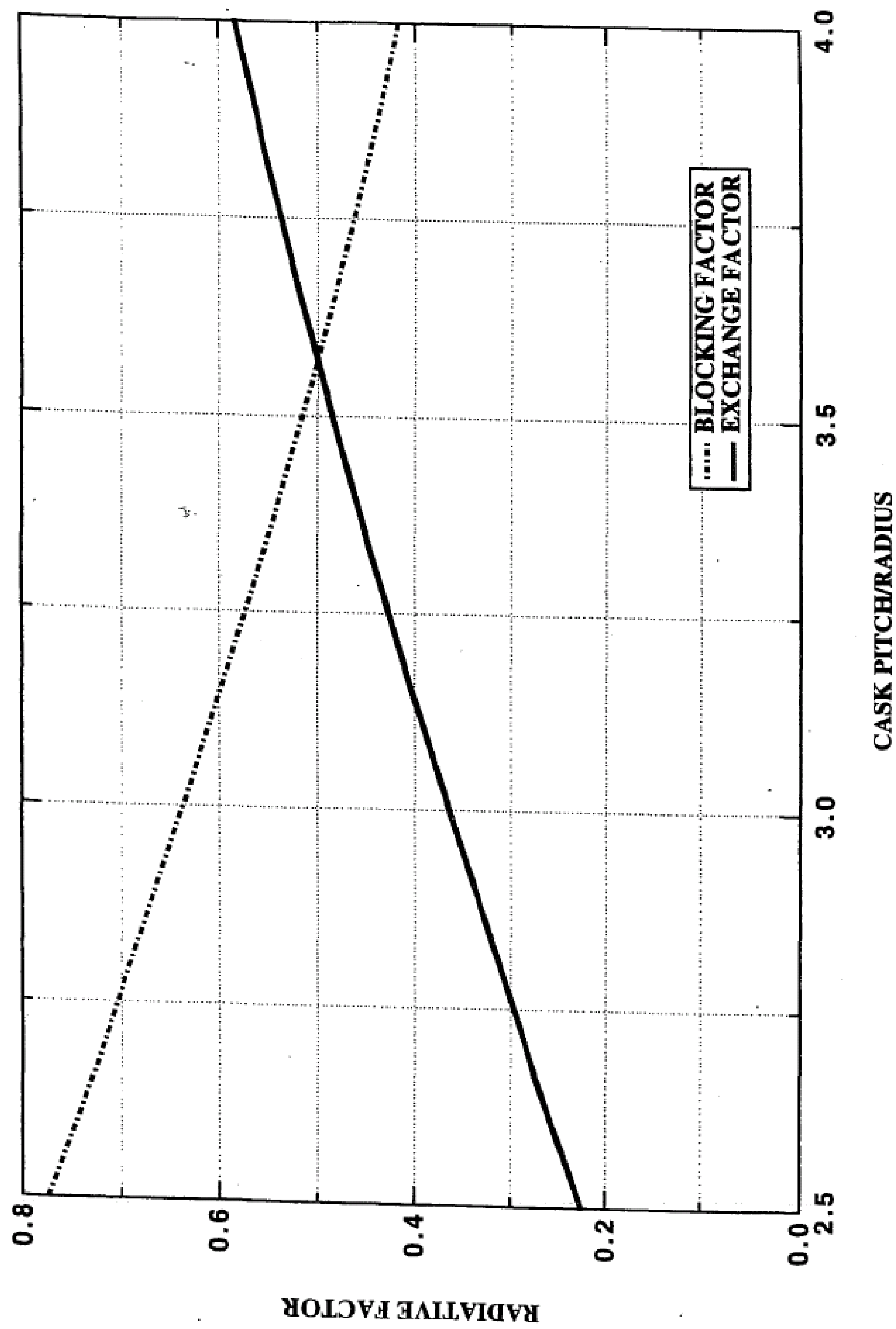

FIGURE 4.4.6; EFFECT OF ISFSI CASK ARRAY PITCH ON RADIATIVE BLOCKING AND EXCHANGE FACTORS

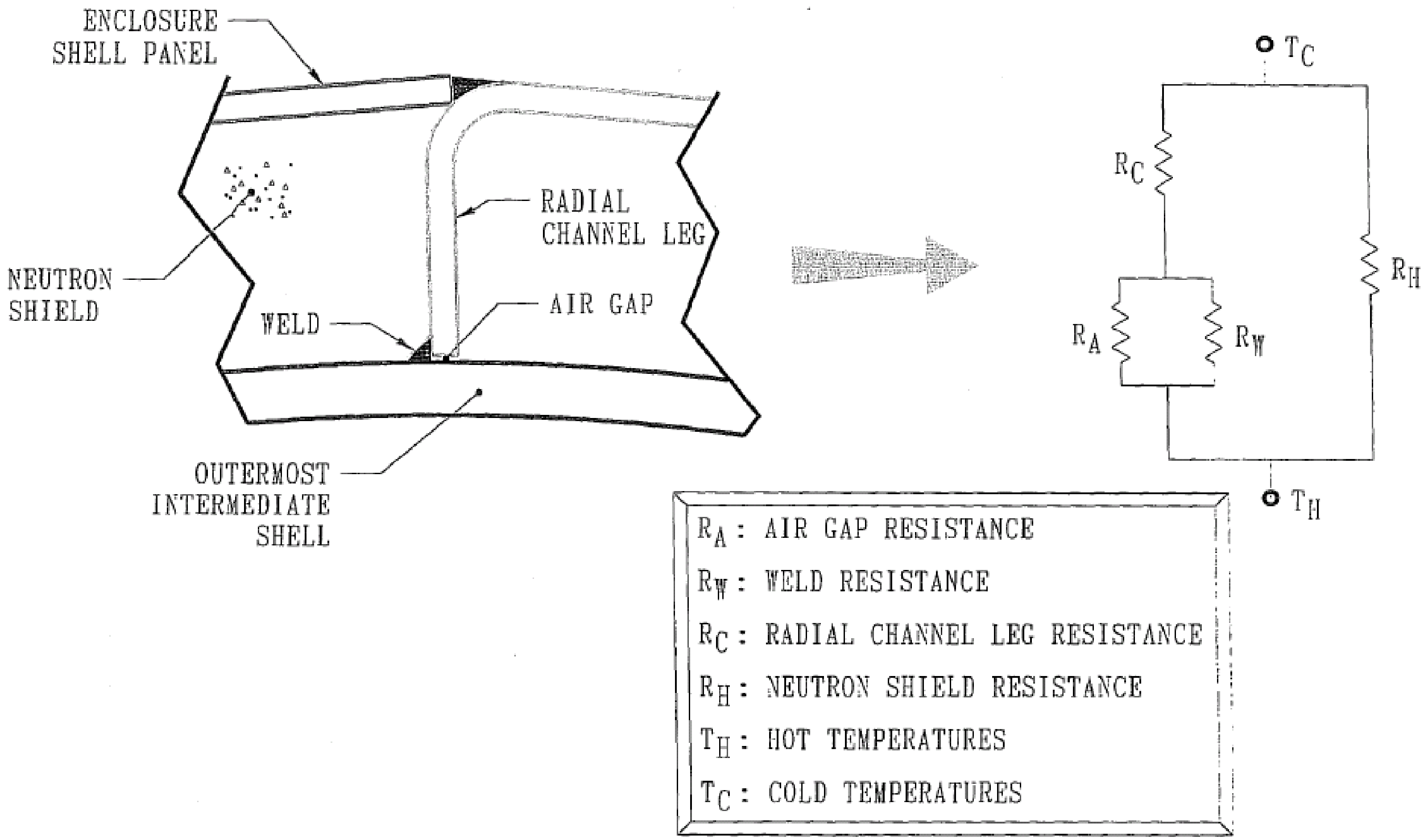

FIGURE 4.4.7; NEUTRON SHIELD REGION RESISTANCE NETWORK ANALOGY FOR EFFECTIVE CONDUCTIVITY CALCULATION

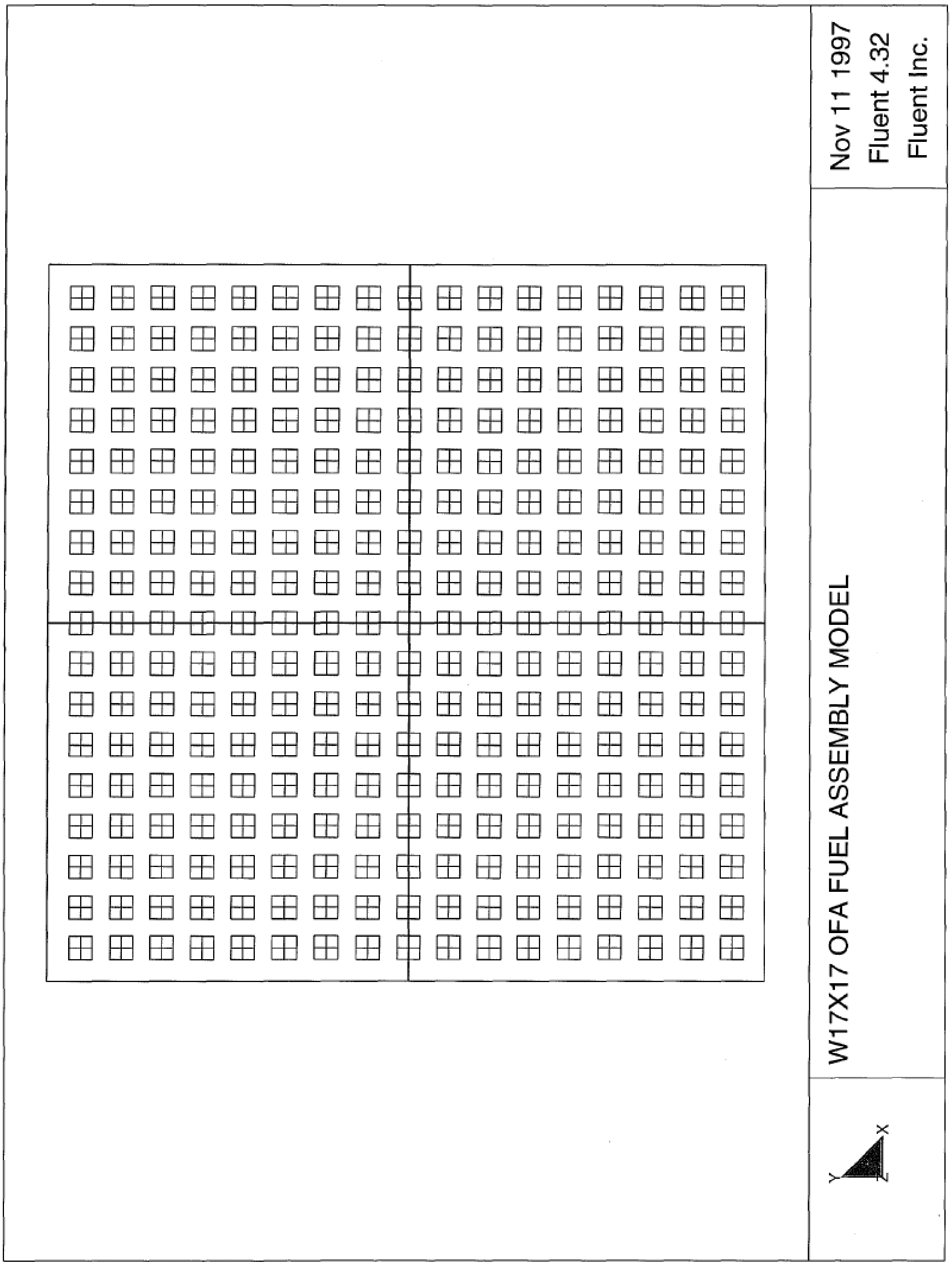

FIGURE 4.4.8: WESTINGHOUSE 17x17 OFA PWR FUEL ASSEMBLY MODEL

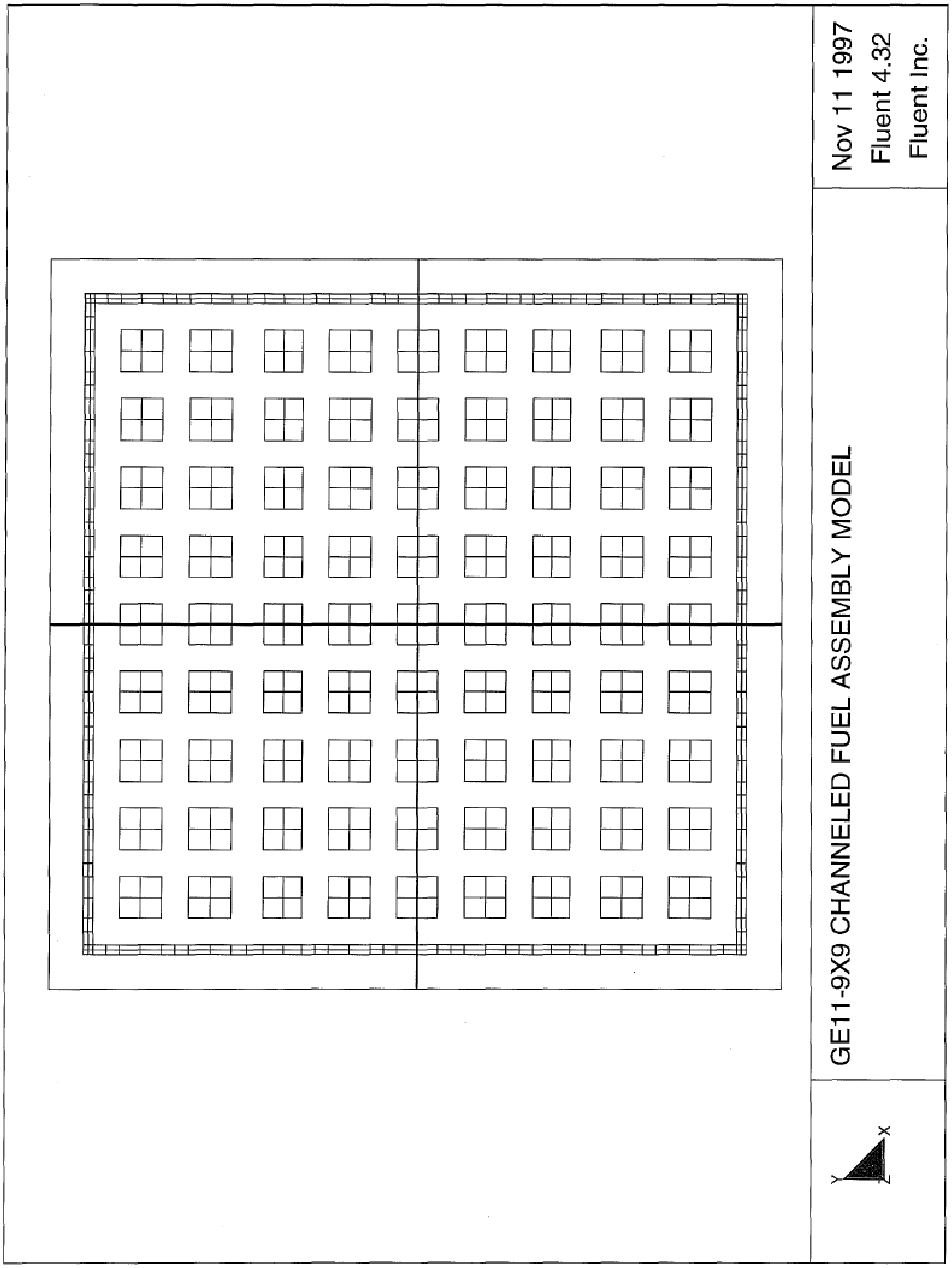

FIGURE 4.4.9: GENERAL ELECTRIC 9x9 BWR FUEL ASSEMBLY MODEL

FIGURE 4.4.10

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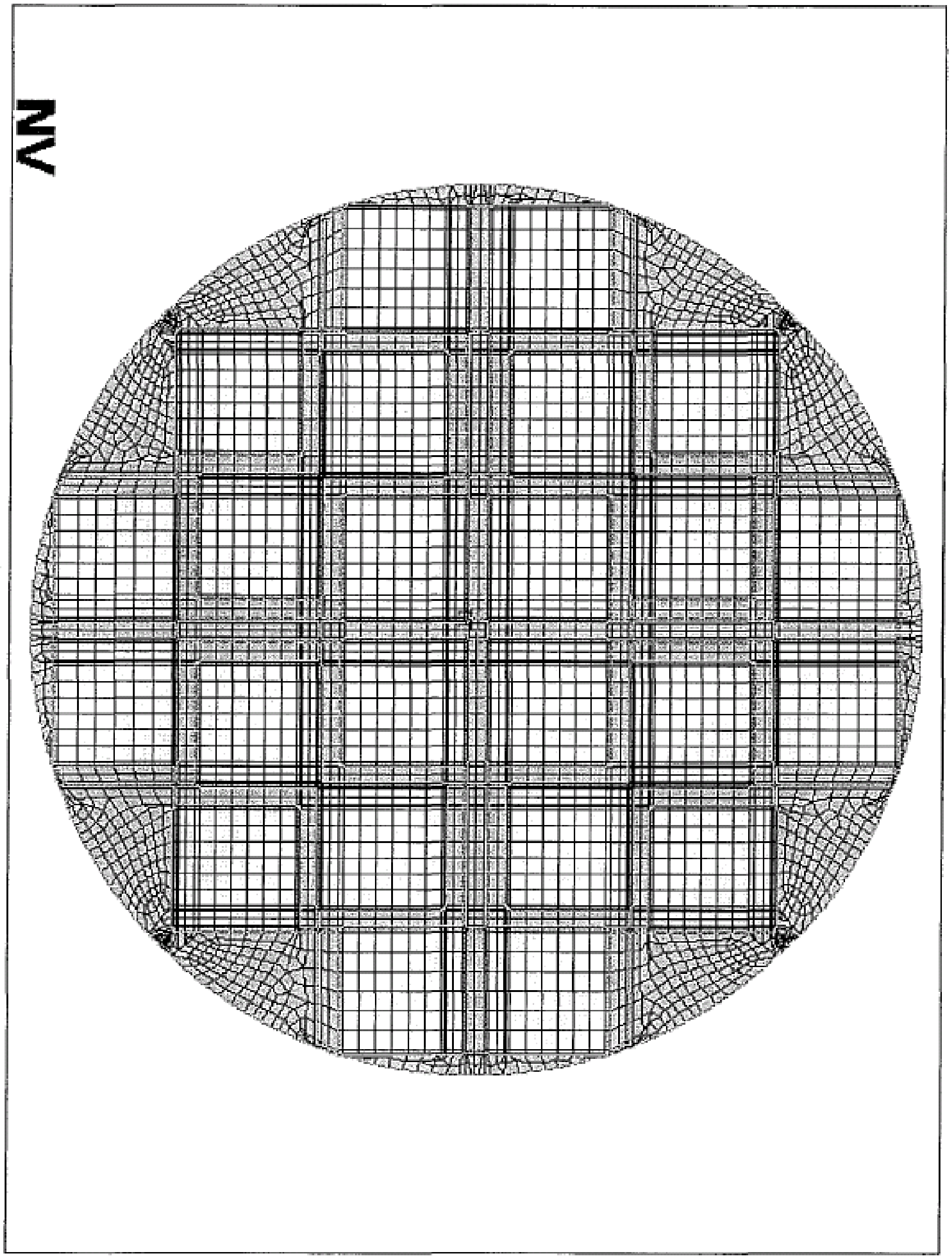


FIGURE 4.4.11; MPC-24 BASKET CROSS-SECTION ANSYS FINITE-ELEMENT MODEL

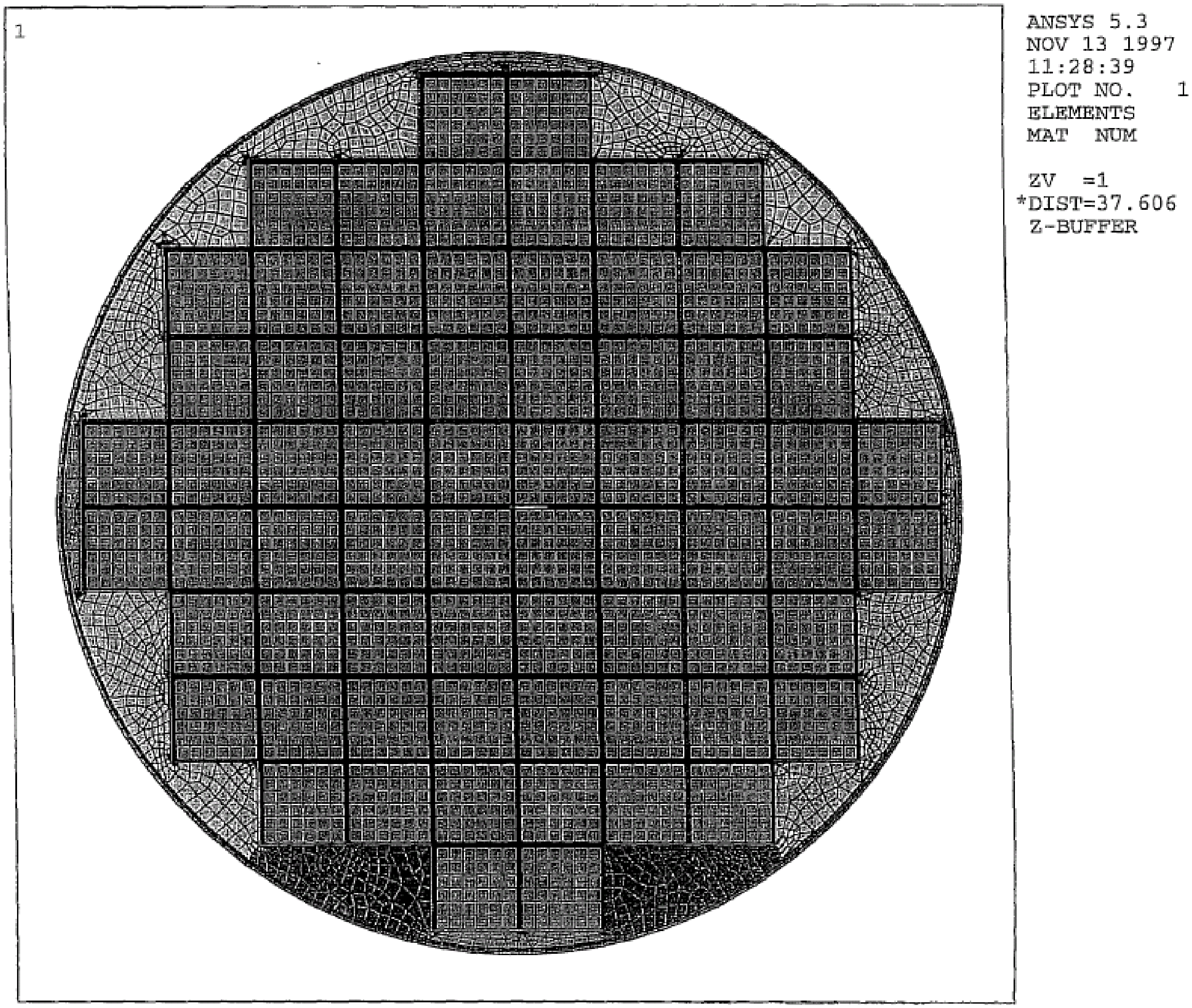


FIGURE 4.4.12; MPC-68 BASKET CROSS-SECTION ANSYS FINITE ELEMENT MODEL

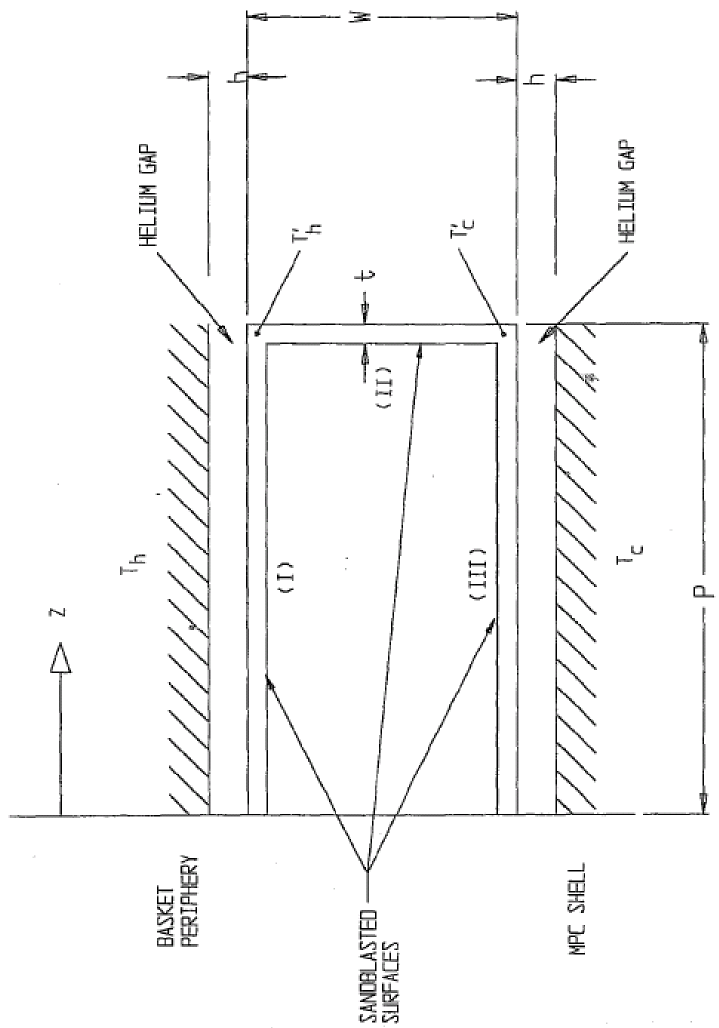
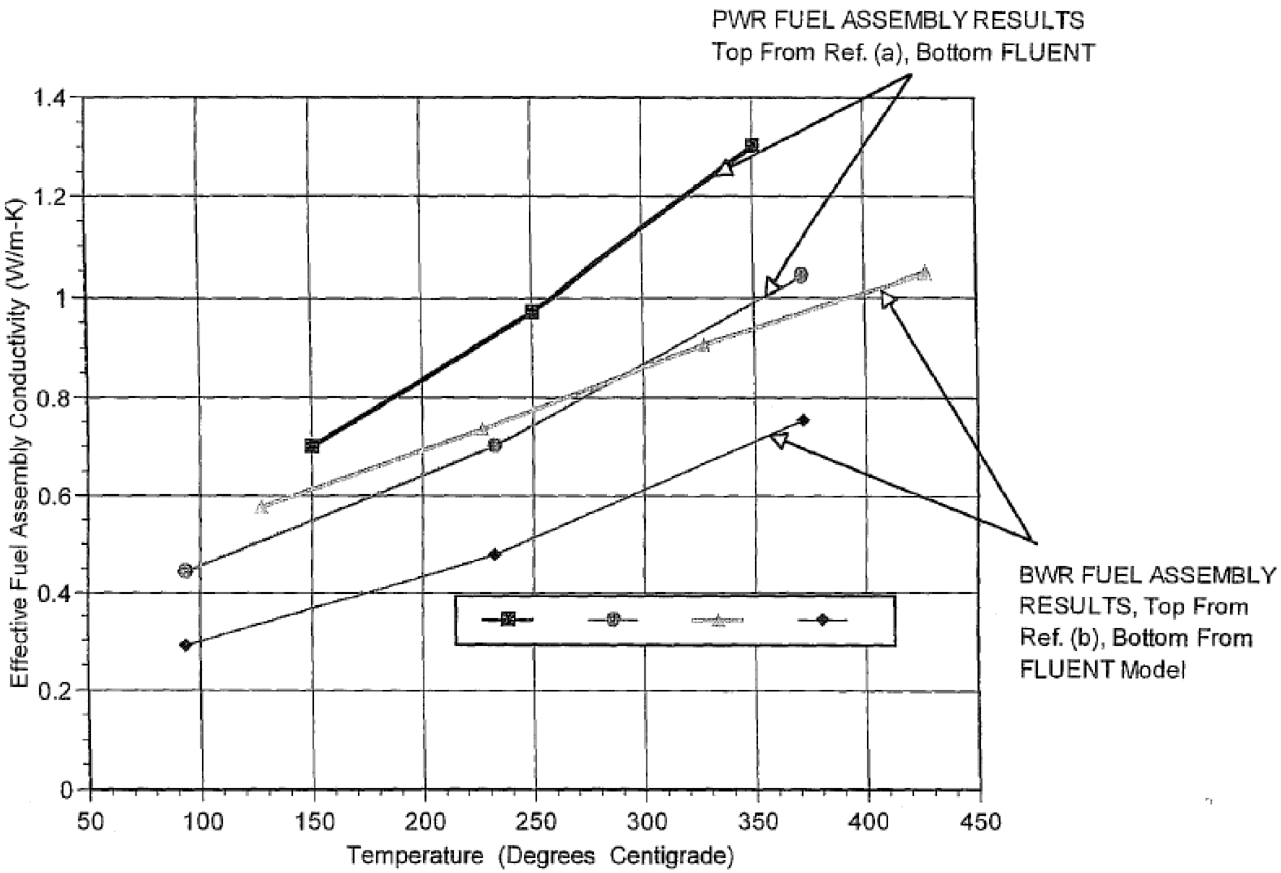


FIGURE 4.4.13; ILLUSTRATION OF AN MPC BASKET TO SHELL ALUMINUM HEAT CONDUCTION ELEMENT



(a) "<u>Determination of SNF Peak Temperatures in the Waste Package</u>", Bahney & Doering, *HLRWM Sixth Annual Conf.*, Pages 671-673, (April 30 - May 5, 1995)
(b) "<u>A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements</u>", *Sandia Report SAND90-2406*, page II-132, (1992)

FIGURE 4.4.14: COMPARISON OF FLUENT BASED FUEL ASSEMBLY EFFECTIVE CONDUCTIVITY RESULTS WITH PUBLISHED TECHNICAL DATA

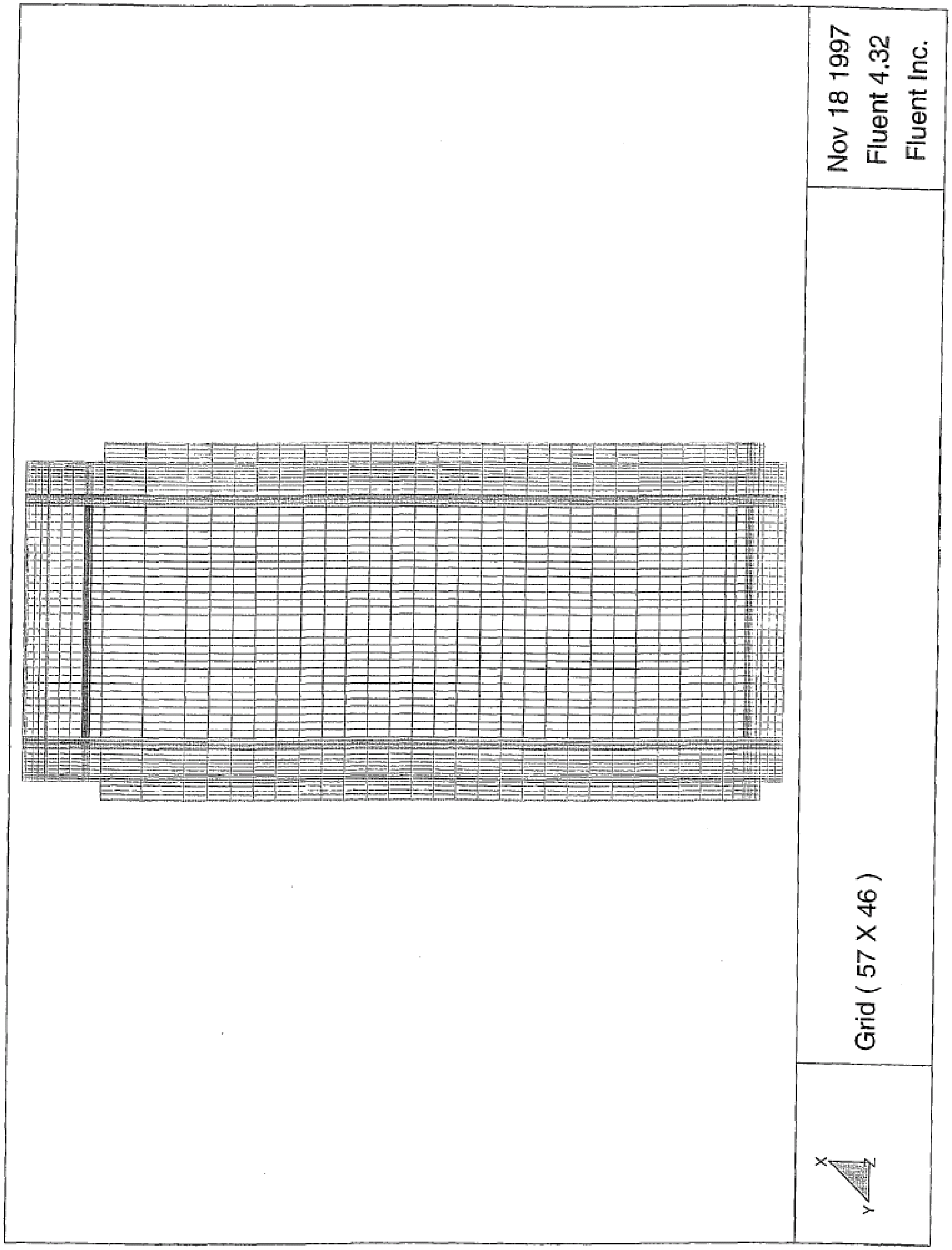


FIGURE 4.4.15: TYPICAL HI-STAR 100 SYSTEM FINITE ELEMENT MESH FOR THERMAL ANALYSIS

FIGURE 4.4.16

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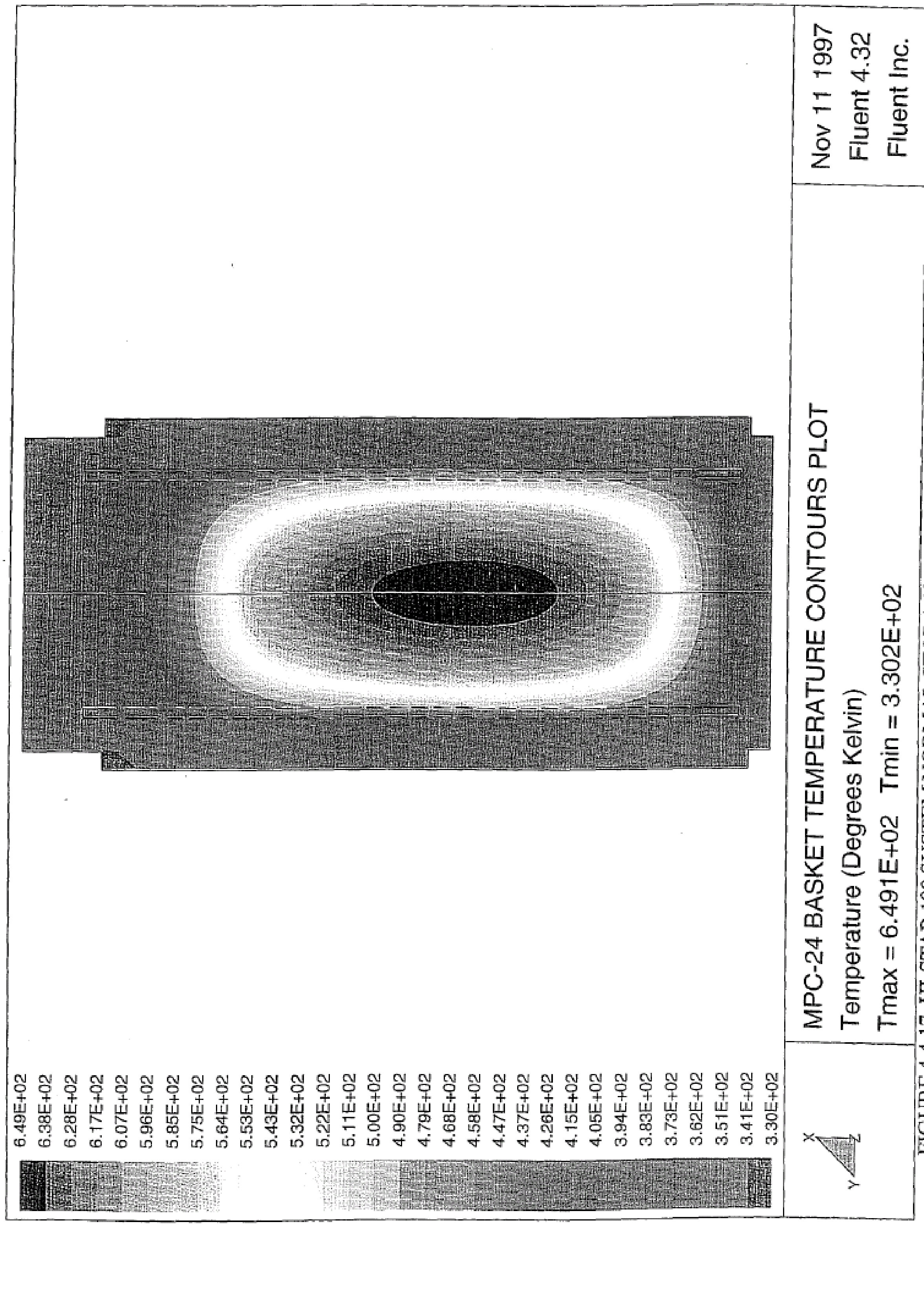

FIGURE 4.4.17: HI-STAR 100 SYSTEM NORMAL STORAGE CONDITION TEMPERATURE CONTOURS PLOT
(MPC-24 BASKET)

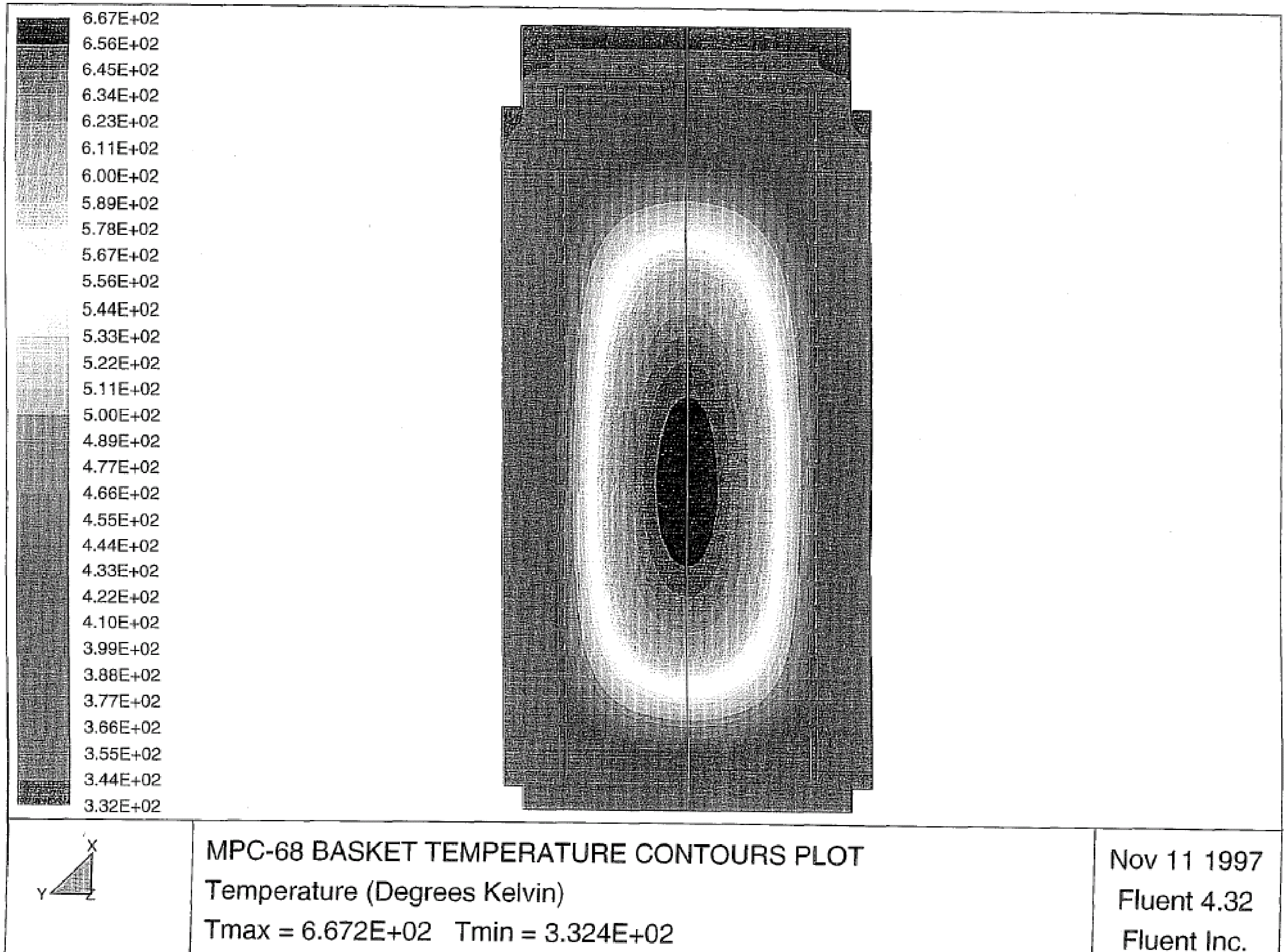

FIGURE 4.4.18: HI-STAR 100 SYSTEM NORMAL STORAGE CONDITION TEMPERATURE CONTOURS PLOT (MPC-68 BASKET)

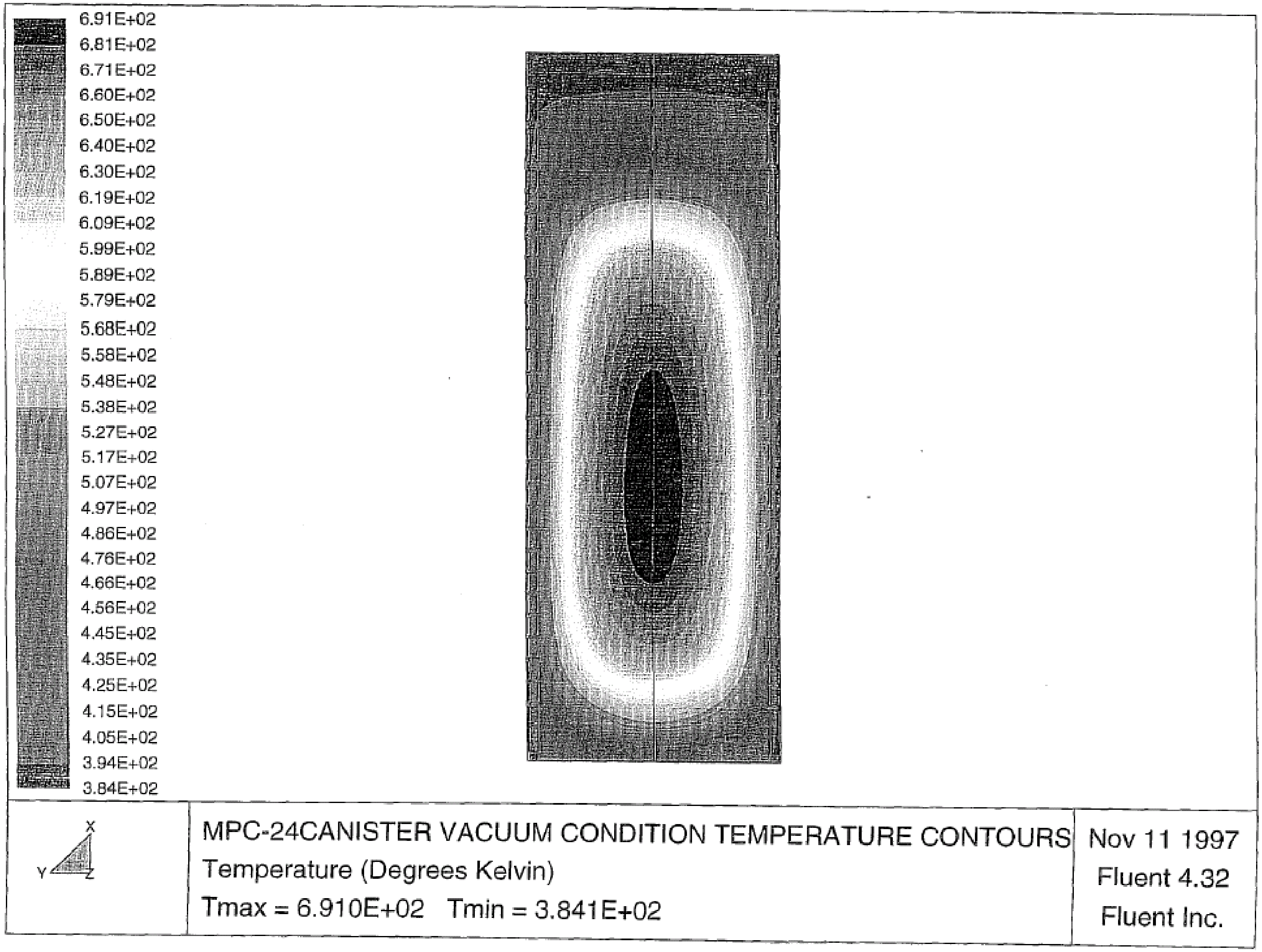

FIGURE 4.4.19: VACUUM CONDITION TEMPERATURE CONTOURS PLOT FOR BOUNDING MPC-24 BASKET

FIGURE 4.4.20

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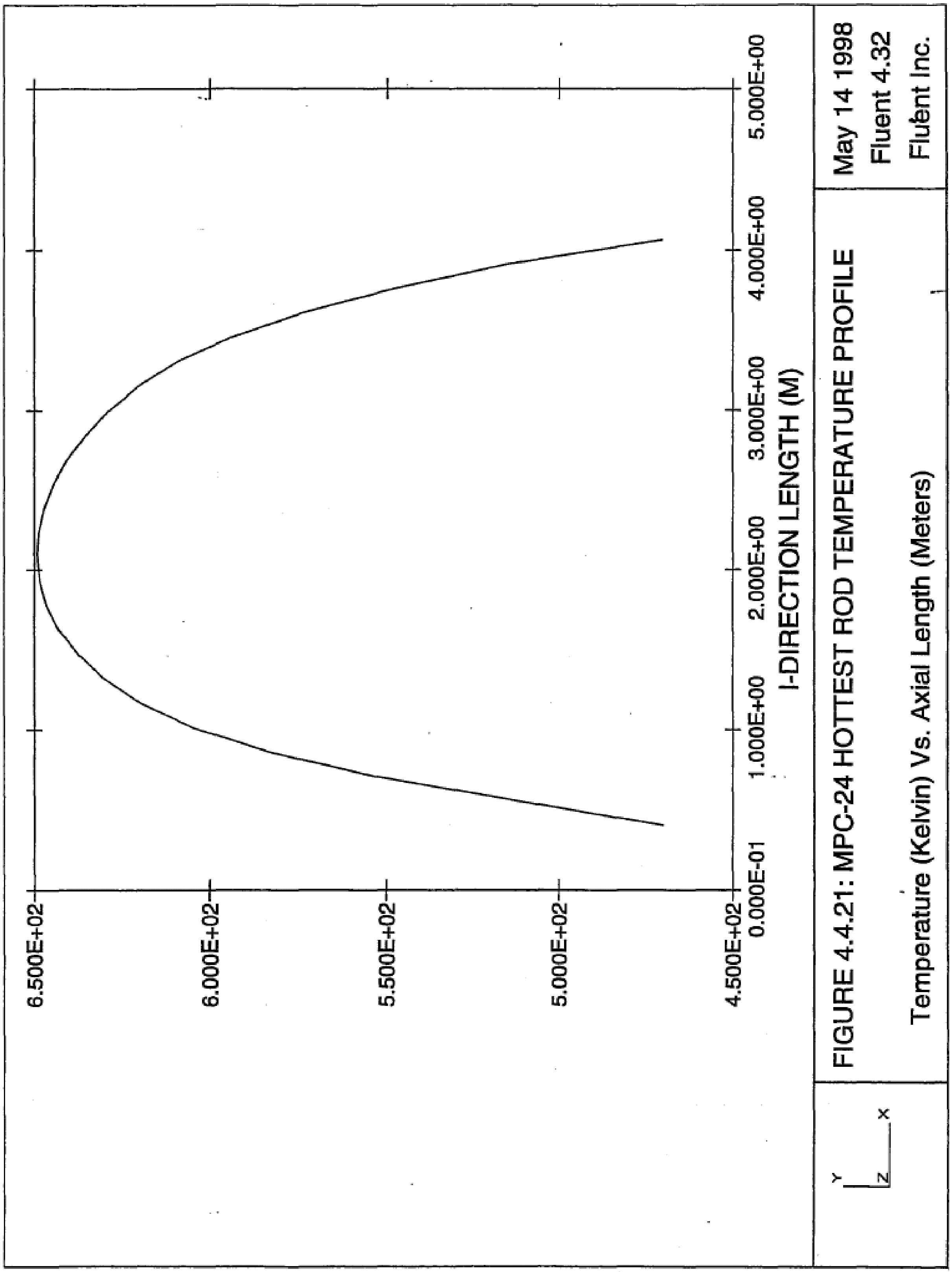

FIGURE 4.4.21: MPC-24 HOTTEST ROD TEMPERATURE PROFILE

Temperature (Kelvin) Vs. Axial Length (Meters)

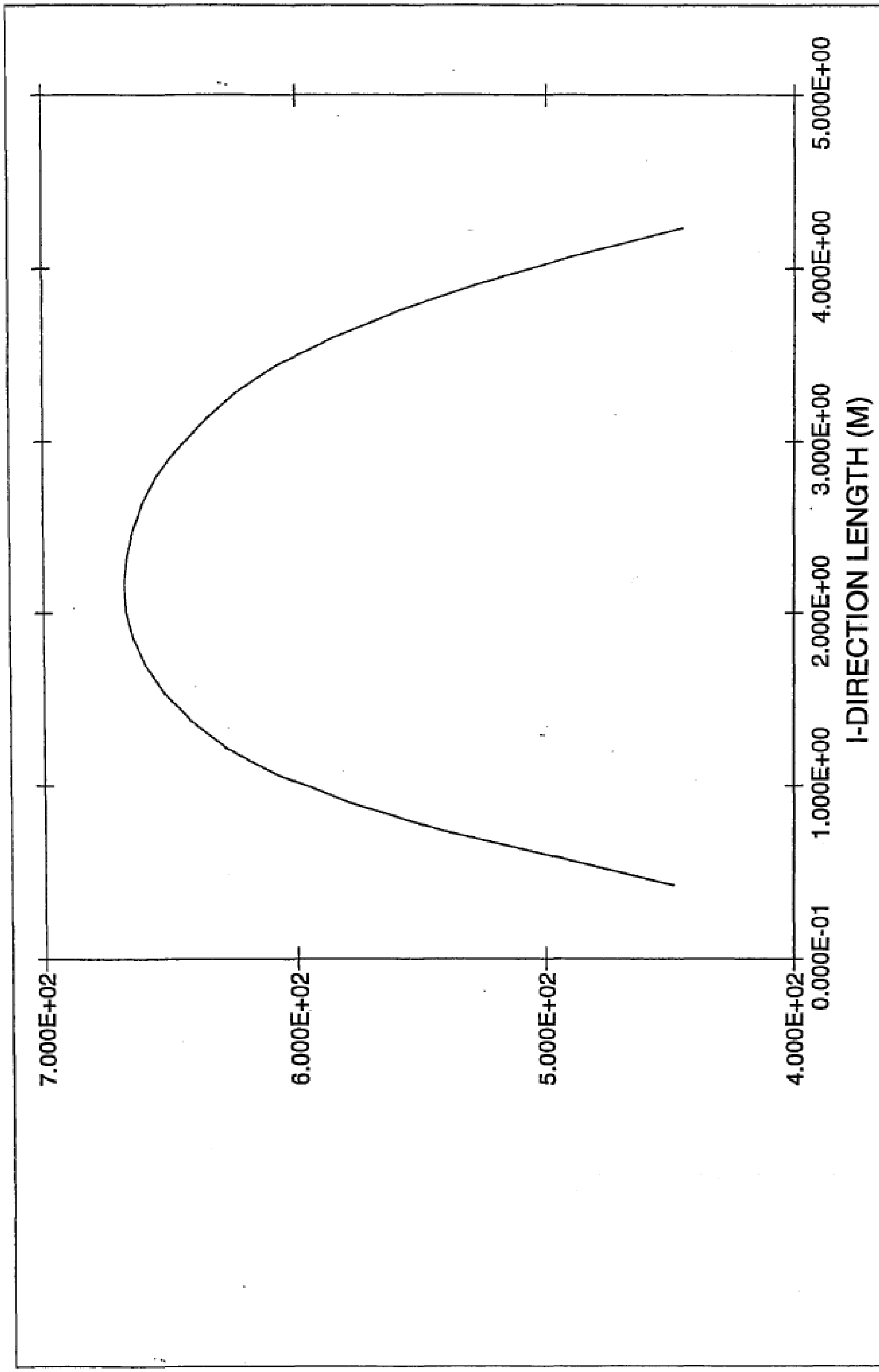

FIGURE 4.4.22: MPC-68 HOTTEST ROD TEMPERATURE PROFILE

Temperature (Kelvin) Vs. Axial Length (Meters)

May 14 1998
Fluent 4.32
Fluent Inc.

FIGURE 4.4.23

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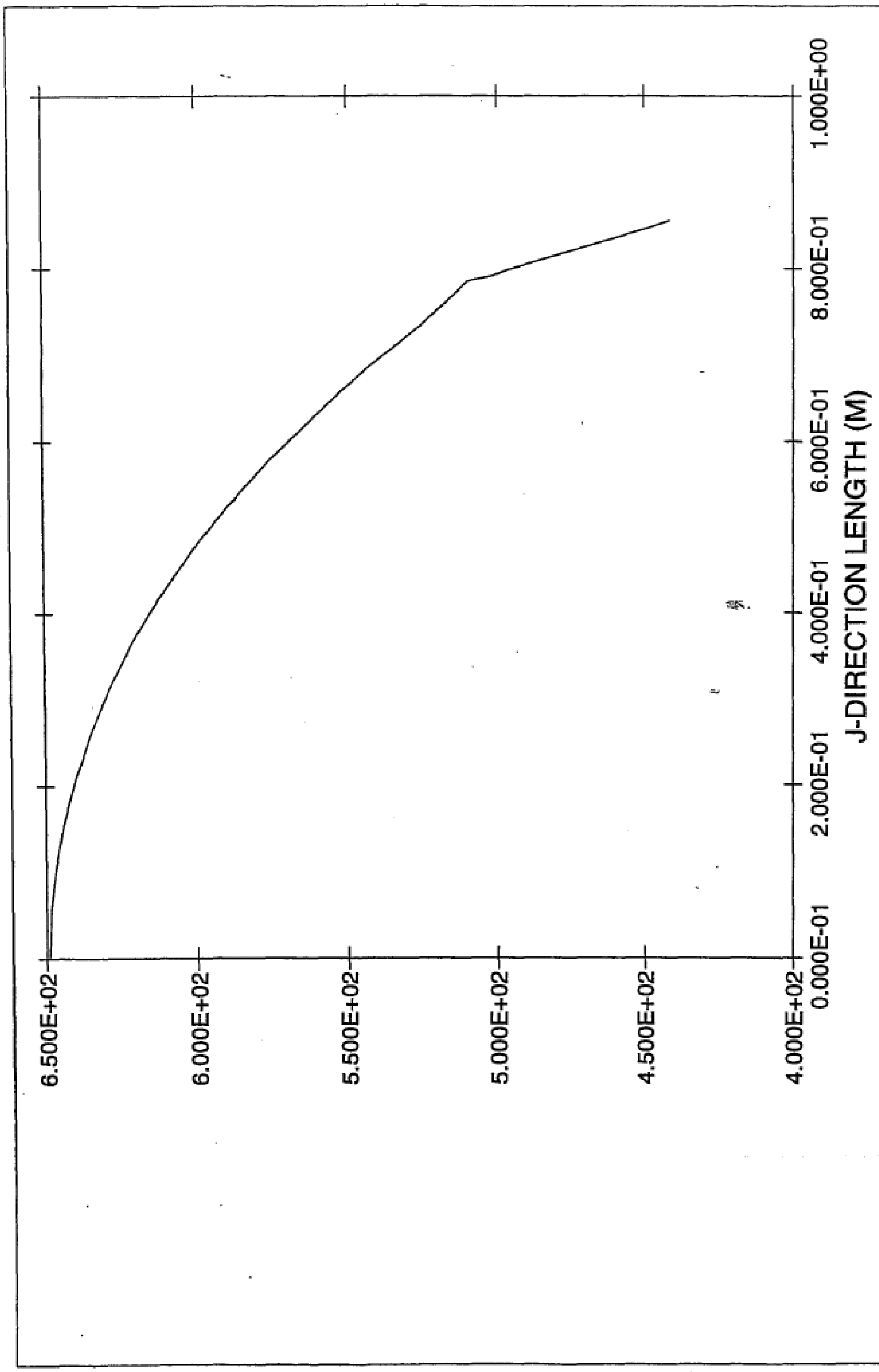

FIGURE 4.4.24: MPC-24 BASKET RADIAL TEMPERATURE PROFILE
(Hottest Basket Cross-Section)
Temperature (Kelvin) Vs. Radial Distance (Meters)

May 14 1998
Fluent 4.32
Fluent Inc.

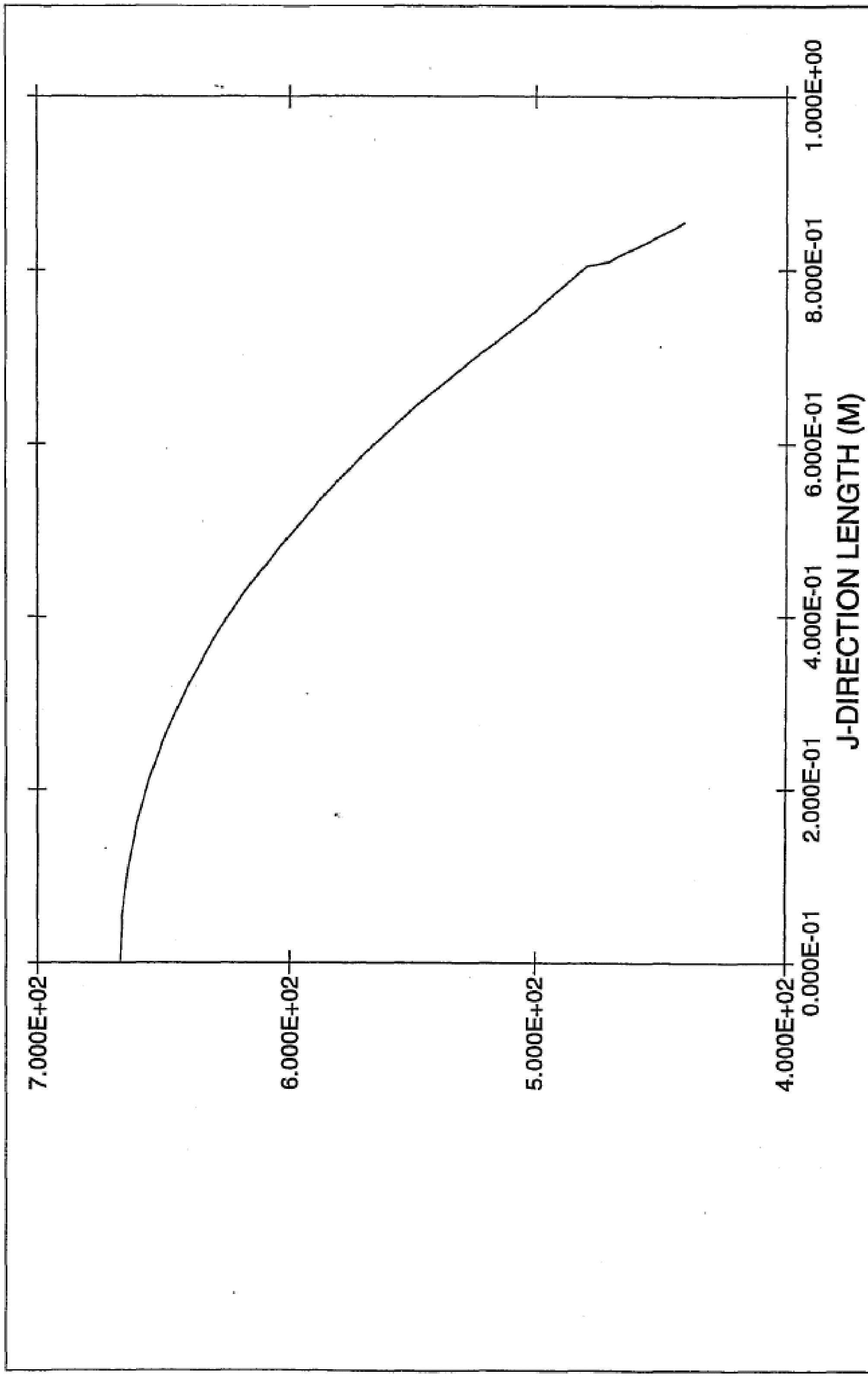

FIGURE 4.4.25: MPC-68 BASKET RADIAL TEMPERATURE PROFILE

Temperature (Kelvin) Vs. Radial Position (Meters)

May 14 1998
Fluent 4.32
Fluent Inc.

4.5 REGULATORY COMPLIANCE

NUREG-1536 ([4.1.3], IV) and ISG-11 [4.3.8] defines thermal acceptance criteria which are addressed in Sections 4.1 through 4.4. Each of the pertinent criteria and the conclusion of the evaluations are summarized here.

1. As required by ISG-11 [4.3.8], the fuel cladding temperature is maintained below the anticipated damage-threshold temperatures for normal conditions for the licensed life of the HI-STAR System. Maximum clad temperatures for normal storage conditions are reported in Paragraph 4.4.2.1.

2. As required by ISG-11 [4.3.8], the fuel cladding temperature is maintained below 570ºC (1058ºF) for short-term accident conditions and short-term off-normal conditions. Results of off-normal and accident condition evaluations presented in Chapter 11 comply with this limit. Maximum clad temperatures for vacuum drying conditions are reported in Paragraph 4.4.2.2 which comply within this limit by large conservative margins. For fuel transfer operations, the fuel cladding temperature is maintained below 570ºC (1058ºF) for MBF and below 400ºC (752ºF) for HBF.

3. As required by NUREG-1536 ([4.1.3], 4.IV.3), the maximum internal pressure of the cask remains within its design pressure for normal, off-normal, and accident conditions, assuming rupture of 1 percent, 10 percent, and 100 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods. Maximum internal pressures are reported in Subsection 4.4.4. Design pressures are summarized in Table 2.2.1.

4. As required by NUREG-1536 ([4.1.3], 4.IV.4), all cask and fuel materials are maintained within their minimum and maximum temperatures for normal and off-normal conditions in order to enable components to perform their intended safety functions. During normal fuel handling operations (e.g., vacuum drying) the cask component temperatures are compared with short-term temperature limits (Paragraph 4.4.2.2). Maximum and minimum temperatures for normal conditions are reported in Subsections 4.4.2 and 4.4.3, respectively. Design temperature limits are summarized in Table 2.2.3. HI-STAR 100 System components defined as important to safety are listed in Table 2.2.6. Off-normal and accident condition thermal evaluations are discussed in Sections 11.1 and 11.2, respectively.

5. & 6. As required by NUREG-1536 ([4.1.3], 4.IV.5), the cask system ensures a very low probability of cladding breach during long-term storage. For long-term normal conditions, the maximum CSF cladding temperature is below the ISG-11 [4.3.8] limit of 400°C (752°F). Maximum fuel clad temperature limits are summarized in Table 2.2.3.

7. As required by NUREG-1536 ([4.1.3], 4.IV.7), the cask system is passively cooled. All heat rejection mechanisms described in this chapter, including conduction, natural convection, and thermal radiation, are completely passive.

8. As required by NUREG-1536 ([4.1.3], 4.IV.8), the thermal performance of the cask is within the allowable design criteria specified in FSAR Chapter 2 for normal storage and fuel handling conditions. During normal fuel handling operations (e.g., vacuum drying) the fuel cladding temperatures are compared with short-term temperature limits. All thermal results reported in this chapter are within the design criteria allowable ranges for all normal storage and fuel handling conditions. Off-normal and fire accident condition responses are reported in Section 11.1 and 11.2, respectively.

Finally, the acceptance criteria set forth in NUREG-1536 ([4.1.3], 4.VI) can be demonstrated to have been satisfied on the strength of information provided in this FSAR. Specifically, it is noted that:

- Structures, systems, and components (SSCs) important to safety are described in sufficient detail in Chapters 1, 2 and 4 of this FSAR to enable evaluations of their thermal effectiveness. Cask SSCs important to safety remain within their operating temperature ranges.

- The HI-STAR 100 System is designed with a heat-removal capability having verifiability and reliability consistent with its importance to safety. The cask is designed to provide adequate heat removal capacity without active cooling systems.

- The spent fuel cladding is protected against degradation leading to gross ruptures by maintaining the cladding temperature in an inert helium environment below 752°F (400°C) in accordance with ISG-11 [4.3.8]. Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.

It is therefore concluded that the thermal design of the HI-STAR 100 System is in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the HI-STAR 100 System will allow safe storage of spent fuel for its design life. This finding is reached on the basis of the technical data presented in this FSAR in conjunction with provisions of 10 CFR Part 72, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

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APPENDIX 4.A: THE FORCED HELIUM DEHYDRATION (FHD) SYSTEM

4.A.1 System Overview

The Forced Helium Dehydration (FHD) system is used to remove the remaining moisture in the MPC cavity after all of the water that can practically be removed through the drain line using a hydraulic pump or an inert gas has been expelled in the water blowdown operation. Expelling the water from the MPC using a conventional pump or a water displacement method using inert gas would remove practically all of the contained water except for the small quantity remaining on the MPC baseplate below the bottom of the drain line and an even smaller adherent amount wetting the internal surfaces. A skid-mounted, closed loop dehydration system will be used to remove the residual water from the MPC such that the partial pressure of the trace quantity of water vapor in the MPC cavity gas is brought down to ≤ 3 torr. The FHD system, engineered for this purpose, shall utilize helium gas as the working substance.

The FHD system, schematically illustrated in Figure 4.A.1, can be viewed as an assemblage of four thermal modules, namely, (i) the condensing module, (ii) the demoisturizer module, (iii) the helium circulator module and (iv) the pre-heater module. The condensing module serves to cool the helium/vapor mixture exiting the MPC to a temperature well below its dew point such that water may be extracted from the helium stream. The condensing module is equipped with suitable instrumentation to provide a direct assessment of the extent of condensation that takes place in the module during the operation of the FHD system. The demoisturizer module, engineered to receive partially cooled helium exiting the condensing module, progressively chills the recirculating helium gas to a temperature that is well below the temperature corresponding to the partial pressure of water vapor at 3 torr.

The motive energy to circulate helium is provided by the helium circulator module, which is sized to provide the pressure rise necessary to circulate helium at the requisite rate. The last item, labeled the pre-heater module, serves to pre-heat the flowing helium to the desired temperature such that it is sufficiently warm to boil off any water present in the MPC cavity.

The pre-heater module, in essence, serves to add supplemental heat energy to the helium gas (in addition to the heat generated by the stored SNF in the MPC) so as to facilitate rapid conversion of water into vapor form. The heat input from the pre-heater module can be adjusted in the manner of a conventional electric heater so that the recirculating helium entering the MPC is sufficiently dry and hot to evaporate water, but not unduly hot to place unnecessary thermal burden on the condensing module.

The FHD system described in the foregoing performs its intended function by continuously removing water entrained in the MPC through successive cooling, moisture removal and reheating of the working substance in a closed loop. In a classical system of the FHD genre, the moisture removal operation occurs in two discrete phases. In the beginning of the FHD system's operation (Phase 1), the helium exiting the MPC is laden with water vapor produced by boiling of the entrained bulk water. The condensing module serves as the principal device to condense out the water vapor from the helium stream in Phase 1. Phase 1 ends when all of the bulk water in the MPC cavity is vaporized. At this point, the operation of the FHD system moves on to steadily lowering the relative humidity and bulk temperature of the circulating helium gas (Phase 2). The demoisturizer module, equipped with the facility to chill flowing helium, plays the principal role in the dehydration process in Phase 2.

4.A.2 Design Criteria

The design criteria set forth below are intended to ensure that design and operation of the FHD system will drive the partial pressure of the residual vapor in the MPC cavity to ≤ 3 torr if the temperature of helium exiting the demoisturizer has met the value and duration criteria provided in the HI-STAR 100 technical specifications. The FHD system shall be designed to ensure that during normal operation (i.e., excluding startup and shutdown ramps) the following criteria are met:

i. The temperature of helium gas in the MPC shall be at least 15ºF higher than the saturation temperature at coincident pressure.

ii. The pressure in the MPC cavity space shall be less than or equal to 60.3 psig (75 psia).

iii. The recirculation rate of helium shall be sufficiently high (minimum hourly throughput equal to ten times the nominal helium mass backfilled into the MPC for fuel storage operations) so as to produce a turbulated flow regime in the MPC cavity.

iv. The partial pressure of the water vapor in the MPC cavity will not exceed 3 torr if the helium temperature at the demoisturer outlet is ≤ 21ºF for a period of 30 minutes.

In addition to the above system design criteria, the individual modules shall be designed in accordance with the following criteria:

i. The condensing module shall be designed to de-vaporize the recirculating helium gas to a dew point of 120ºF or less.

ii. The demoisturizer module shall be configured to be introduced into its helium conditioning function <u>after</u> the condensing module has been operated for the required length of time to assure that the bulk moisture vaporization in the MPC has been completed.

iii. The helium circulator shall be sized to effect the minimum flow rate of circulation required by the system design criteria described above.

iv. The pre-heater module shall be engineered to ensure that the temperature of the helium gas in the MPC meets the system design criteria described above.

4.A.3 Analysis Requirements

The design of the FHD system shall be subject to the confirmatory analyses listed below to ensure that the system will accomplish the performance objectives set forth in this FSAR.

i. System thermal analysis in Phase 1: Characterize the rate of condensation in the condensing module and helium temperature variation under Phase 1 operation (i.e., the scenario where there is some unevaporated water in the MPC) using a classical thermal-hydraulic model wherein the incoming helium is assumed to fully mix with the moist helium inside the MPC.

ii. System thermal analysis in Phase 2: Characterize the thermal performance of the closed loop system in Phase 2 (no unvaporized moisture in the MPC) to predict the rate of condensation and temperature of the helium gas exiting the condensing and the demoisturizer modules. Establish that the system design is capable to ensure that partial pressure of water vapor in the MPC will reach ≤ 3 torr if the temperature of the helium gas exiting the demoisturizer is predicted to be at a maximum of 21ºF for 30 minutes.

iii. Fuel Cladding Temperature Analysis: A steady-state thermal analysis of the MPC under the forced helium flow scenario shall be performed using the methodology described in Subsections 4.4.1.1.1 through 4.4.1.1.4 with due recognition of the forced convection process during FHD system operation. This analysis shall demonstrate that the peak temperature of the fuel cladding under the most adverse condition of FHD system operation (design maximum heat load, no moisture, and maximum helium inlet temperature), is below the peak cladding temperature limit for normal conditions of storage for the applicable fuel type (PWR or BWR) and cooling time at the start of dry storage.

4.A.4 Acceptance Testing

The first FHD system designed and built for the MPC drying function required by technical specifications shall be subject to confirmatory testing as follows:

a. A representative quantity of water shall be placed in a manufactured MPC (or equivalent mock-up) and the closure lid and RVOAs installed and secured to create a hermetically sealed container.

b. The MPC cavity drying test shall be conducted for the worst case scenario (no heat generation within the MPC available to vaporize water).

c. The drain and vent line RVOAs on the MPC lid shall be connected to the terminals located in the pre-heater and condensing modules of the FHD system, respectively.

d. The FHD system shall be operated through the moisture vaporization (Phase 1) and subsequent dehydration (Phase 2). The FHD system operation will be stopped after the temperature of helium exiting the demoisturizer module has been at or below 21ºF for thirty minutes (nominal). Thereafter, a sample of the helium gas from the MPC will be extracted and tested to determine the partial pressure of the residual water vapor in it. The FHD system will be deemed to have passed the acceptance testing if the partial pressure in the extracted helium sample is less than or equal to 3 torr.

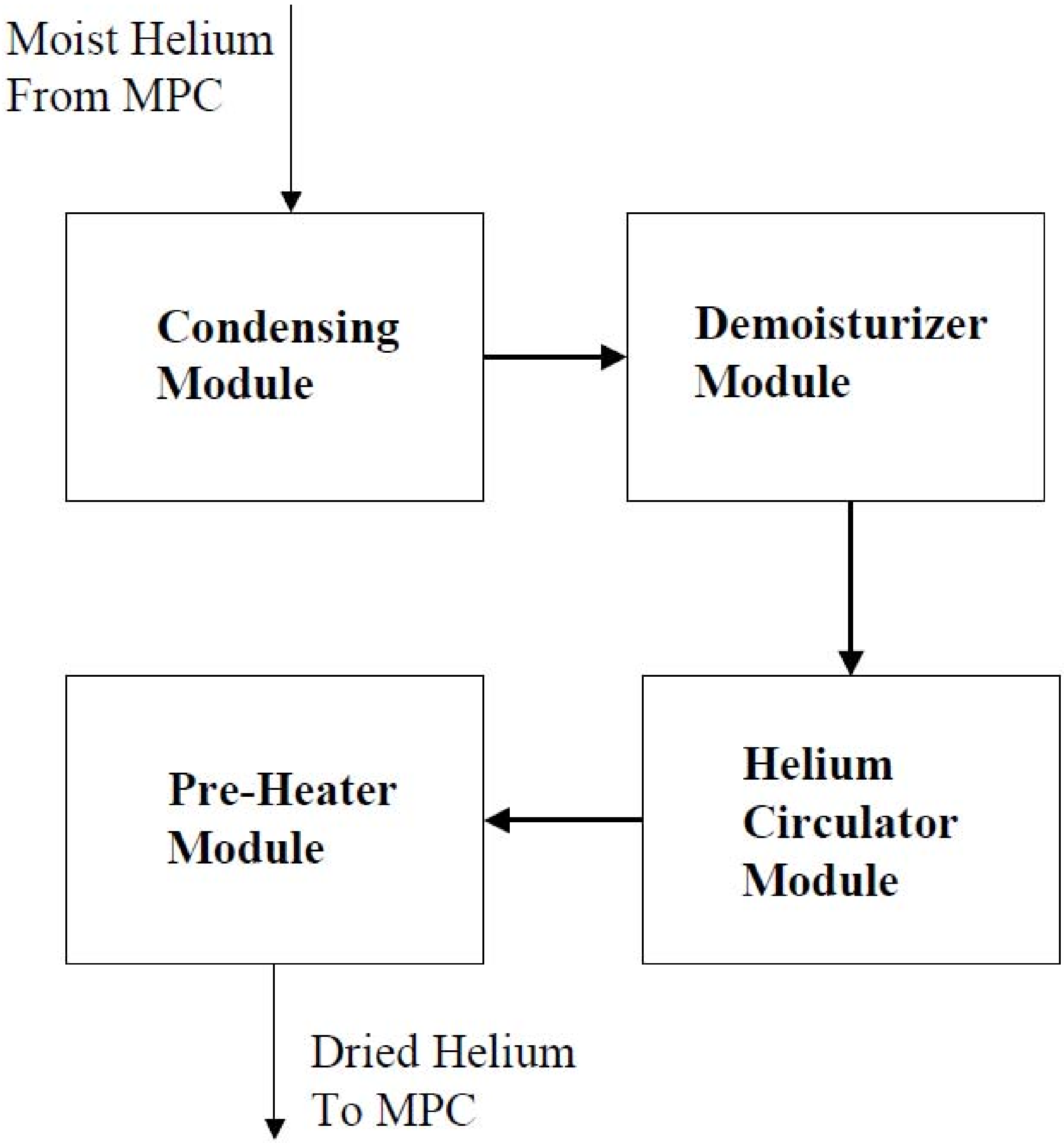


FIGURE 4.A.1: SCHEMATIC OF THE FORCED HELIUM DEHYDRATION SYSTEM

CHAPTER 5: SHIELDING EVALUATION

5.0 INTRODUCTION

The shielding analysis of the HI-STAR 100 System is presented in this chapter. The HI-STAR 100 System is designed to accommodate different MPCs within one standard HI-STAR overpack. The MPCs are designated as MPC-24 (24 PWR fuel assemblies), MPC-32 (32 PWR fuel assemblies) and MPC-68 (68 BWR fuel assemblies).

In addition to storing intact PWR and BWR fuel assemblies, the HI-STAR 100 System is designed to store damaged BWR fuel assemblies and BWR fuel debris. Damaged fuel assemblies and fuel debris are defined in Section 2.1.3. Both damaged BWR fuel assemblies and BWR fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. DFCs containing fuel debris must be stored in the MPC-68F. DFCs containing damaged fuel assemblies may be stored in either the MPC-68 or the MPC-68F.

The MPC-68 and MPC-68F are also capable of storing Dresden Unit 1 antimony-beryllium neutron sources and the single Thoria rod canister which contains 18 thoria rods that were irradiated in two separate fuel assemblies.

PWR fuel assemblies may contain burnable poison rod assemblies (BPRAs) or thimble plug devices (TPDs) or similarly named devices. These devices are an integral yet removable part of PWR fuel assemblies and therefore the HI-STAR 100 System has been designed to store PWR fuel assemblies with or without BPRAs or TPDs. Since BPRAs and TPDs occupy the same space within a fuel assembly, a single PWR fuel assembly will not contain both devices.

The sections that follow will demonstrate that the design of the HI-STAR 100 dry cask storage system fulfills the following acceptance criteria outlined in the Standard Review Plan, NUREG-1536 [5.2.1]:

Acceptance Criteria

1. The minimum distance from each spent fuel handling and storage facility to the controlled area boundary must be at least 100 meters. The “controlled area” is defined in 10CFR72.3 as the area immediately surrounding an ISFSI or monitored retrievable storage (MRS) facility, for which the licensee exercises authority regarding its use and within which ISFSI operations are performed.

2. The cask vendor must show that, during both normal operations and anticipated occurrences, the radiation shielding features of the proposed dry cask storage system are sufficient to meet the radiation dose requirements in Sections 72.104(a).

Specifically, the vendor must demonstrate this capability for a typical array of casks in the most bounding site configuration. For example, the most bounding configuration might be located at the minimum distance (100 meters) to the controlled area boundary, without any shielding from other structures or topography.

3. Dose rates from the cask must be consistent with a well-established “as low as reasonably achievable” (ALARA) program for activities in and around the storage site.

4. After a design-basis accident, an individual at the boundary or outside the controlled area shall not receive a dose greater than the limits specified in 10 CFR 72.106.

5. The proposed shielding features must ensure that the dry cask storage system meets the regulatory requirements for occupational and radiation dose limits for individual members of the public, as prescribed in 10 CFR Part 20, Subparts C and D.

This chapter contains the following information which demonstrates full compliance with the Standard Review Plan, NUREG-1536:

- A description of the shielding features of the HI-STAR 100 System.
- A description of the bounding source terms.
- A general description of the shielding analysis methodology.
- A description of the analysis assumptions and results for the HI-STAR 100 System.
- Analyses are presented for each MPC showing that the radiation dose rates follow As-Low-As-Reasonably-Achievable (ALARA) practices.
- The HI-STAR 100 System has been analyzed to show that the 10CFR72.104 and 10CFR72.106 controlled area boundary radiation dose limits are met during normal, off-normal, and accident conditions of storage for non-effluent radiation from illustrative ISFSI configurations at a minimum distance of 100 meters.
- Analyses are also presented which demonstrate that the storage of damaged fuel and fuel debris in the HI-STAR 100 System is bounded by the BWR intact fuel analysis during normal, off-normal, and accident conditions.

Chapter 10, Radiation Protection, contains the following information:

- A discussion of the estimated occupational exposures for the HI-STAR 100 System.
- A summary of the estimated radiation exposure to the public.

Chapter 2 contains a detailed description of structures, systems, and components important to safety.

Chapter 7 contains an analysis of the estimated dose at the controlled area boundary during normal, off-normal, and accident conditions from the release of radioactive materials. Therefore, this chapter only calculates the dose from direct neutron and gamma radiation emanating from the HI-STAR 100 System.

5.1 DISCUSSION AND RESULTS

The principal sources of radiation in the HI-STAR 100 System are:

- Gamma radiation originating from the following sources

    1. Decay of radioactive fission products
    2. Secondary photons from neutron capture in fissile and non-fissile nuclides
    3. Hardware activation products generated during core operations

- Neutron radiation originating from the following sources

    1. Spontaneous fission
    2. α,n reactions in fuel materials
    3. Secondary neutrons produced by fission from subcritical multiplication
    4. γ,n reactions (this source is negligible)
    5. Dresden Unit 1 antimony-beryllium neutron sources

Shielding from gamma radiation is provided by the steel structure of the MPC and overpack. In order for the neutron shielding to be effective, the neutrons must be thermalized and then absorbed in a material of high neutron cross section. In the HI-STAR 100 design, a neutron shielding material, Holtite-A, is used to thermalize the neutrons. Boron carbide, dispersed in the neutron shield, utilizes the high neutron absorption cross section of ^{10}B to absorb the thermalized neutrons.

The shielding analyses were performed with MCNP-4A [5.1.1] from Los Alamos National Laboratory. The source terms for the design basis fuels were calculated with the SAS2H and ORIGEN-S modules from the SCALE 4.3 system [5.1.2, 5.1.3]. A detailed description of the MCNP models and the source term calculations is presented in Sections 5.3 and 5.2, respectively.

The design basis intact zircaloy clad fuel assemblies used for calculating the dose rates presented in this chapter are B&W 15x15 and the GE 7x7, for PWR and BWR fuel types, respectively. The design basis intact 6x6, damaged, and mixed oxide (MOX) fuel assemblies are the GE 6x6. Table 2.1.6 specifies the acceptable intact zircaloy clad fuel characteristics for storage. Table 2.1.7 specifies the acceptable damaged and MOX zircaloy clad fuel characteristics for storage.

The design basis intact stainless steel clad fuels are the WE 15x15 and the A/C 10x10, for PWR and BWR fuel types, respectively. Table 2.1.11 specifies the acceptable fuel characteristics of stainless steel clad fuel for storage.

The MPC-24, MPC-32 and MPC-68 are qualified for storage of SNF with different combinations of maximum burnup levels and minimum cooling times. Tables 2.1.13 through 2.1.15 specify the acceptable maximum burnup levels and minimum cooling times for storage of zircaloy clad fuel in the MPC-24, MPC-32 and the MPC-68. Table 2.1.11 specifies the acceptable maximum burnup levels and minimum cooling times for storage of stainless steel clad fuel. The values in Tables 2.1.11, and 2.1.13 through 2.1.15 were chosen based on an analysis of the maximum decay heat load that could be accommodated within each MPC. The shielding analyses presented in this chapter used the burnup and cooling time combinations listed below which are either equal to or conservatively bound the acceptable burnup levels and cooling times shown in Tables 2.1.11, and 2.1.13 through 2.1.15.

<table>
<tr><td colspan="3">Maximum Burnup and Minimum Cooling Times Analyzed</td></tr>
<tr><td colspan="3">Zircaloy Clad Fuel</td></tr>
<tr><td>MPC-24</td><td>MPC-32</td><td>MPC-68</td></tr>
<tr><td>40,000 MWD/MTU
5 year cooling</td><td>40,000 MWD/MTU
8 year cooling</td><td>35,000 MWD/MTU
5 year cooling</td></tr>
<tr><td>47,500 MWD/MTU
8 year cooling</td><td>45,000 MWD/MTU
11 year cooling</td><td>45,000 MWD/MTU
9 year cooling</td></tr>
<tr><td>N/A</td><td>N/A</td><td>30,000 MWD/MTU
18 year cooling
(6x6 intact, damaged and MOX fuel)</td></tr>
<tr><td colspan="3">Stainless Steel Clad Fuel</td></tr>
<tr><td colspan="2">MPC-24</td><td>MPC-68</td></tr>
<tr><td colspan="2">30,000 MWD/MTU
9 year cooling</td><td>22,500 MWD/MTU
10 year cooling</td></tr>
<tr><td colspan="2">40,000 MWD/MTU
15 year cooling</td><td>N/A</td></tr>
</table>

The dose rates corresponding to the burnup and cooling time combination which resulted in the highest dose rates at the midplane of the cask during normal conditions are reported in this section. Dose rates for each of the combinations are listed in Section 5.4.

5.1.1 Normal and Off-Normal Operations

Chapter 11 discusses the potential off-normal conditions and their effect on the HI-STAR 100 System. None of the off-normal conditions have any impact on the shielding analysis. Therefore, off-normal and normal conditions are identical for the purpose of the shielding evaluation.

The 10CFR72.104 criteria for radioactive materials in effluents and direct radiation during normal operations are:

1. During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area, must not exceed 25 mrem (0.25 mSv)[1] to the whole body, 75 mrem (0.75 mSv) to the thyroid and 25 mrem (0.25 mSv) to any other critical organ.

2. Operational restrictions must be established to meet as low as reasonably achievable objectives for radioactive materials in effluents and direct radiation.

10CFR20 Subparts C and D specify additional requirements for occupational dose limits and radiation dose limits for individual members of the public. Chapter 10 specifically addresses these regulations.

In accordance with ALARA practices, design objective dose rates are established for the HI-STAR 100 in Section 2.3.5.2 as: 130 mrem/hour (1.3 mSv/h) on the radial surface of the overpack, and 375 mrem/hour (3.75 mSv/h) in areas above and below the neutron shield in the radial direction.

The dose rates presented in this section are calculated at 40,000 MWD/MTU and 5-year cooling for the MPC-24, 40,000 MWD/MTU and 8-year cooling for MPC-32, and 35,000 MWD/MTU and 5-year cooling for the MPC-68. Based on a comparison of the normal condition dose rates at the fuel mid-plane for the various burnup and cooling time combinations analyzed, these were chosen as the worst case for the MPC-24, MPC-32 and the MPC-68. Section 5.4 provides a detailed list of dose rates at several cask locations for all burnup and cooling times analyzed.

Figure 5.1.1 identifies the locations of the dose points referenced in the summary tables. The bottom shield shown in this figure is temporary shielding which may be used during on-site horizontal handling operations. Dose Point #7 is located directly below the overpack bottom plate or directly below the bottom shield when it is attached. Dose Points #1, #3, and #4 are not contact doses, but rather, in-air doses at the locations shown. The dose values reported at the locations shown on Figure 5.1.1 are averaged over a region that is approximately 1 foot in width.

Tables 5.1.1 through 5.1.3 provide the maximum dose rates adjacent to the overpack during normal conditions for each of the MPCs. Tables 5.1.4 through 5.1.6 provide the maximum dose rates at one meter from the overpack.

The dose to any real individual at or beyond the controlled area boundary is required to be below 25 mrem (0.25 mSv) per year. The minimum distance to the controlled area boundary is 100 meters from the ISFSI. Only the MPC-24 and MPC-32 were used in the calculation of the dose

[1] Wherever multiple units are shown, the US units are the governing value, and the SI units are shown for information only.

rates at the controlled area boundary. The MPC-24 was chosen because its dose rates are equivalent or greater than the dose rates from the MPC-68 as shown in Tables 5.1.2, 5.1.3, 5.1.5, and 5.1.6. Table 5.1.7 presents the annual dose to an individual from a single cask and various arrays of casks, assuming 100% occupancy (8760 hours). The minimum distance required for the corresponding dose is also listed. These values were calculated for the MPC-24 with a burnup of 40,000 MWD/MTU and a 5-year cooling time, and for the MPC-32 with a burnup of 40,000 MWD/MTU and a 8-year cooling time. It will be shown in Section 5.4.3 that these burnup and cooling time combinations result in the highest offsite dose for the combinations of maximum burnup and minimum cooling time analyzed. It is noted that these data are provided for illustrative purposes only. A detailed site specific evaluation of dose at the controlled area boundary will be performed for each ISFSI in accordance with 10CFR72.212, as stated in Chapter 12, Operating Controls and Limits. The site specific evaluation will consider dose from other portions of the facility and will consider the specifics of the fuel being stored (burnup and cooling time).

Figure 5.1.2 is an annual dose versus distance graph for the configurations of the cask with MPC-24 provided in Table 5.1.7. This curve, which is based on 100% occupancy, is provided for illustrative purposes only and will be re-evaluated on a site-specific basis.

Section 5.2 lists the gamma and neutron sources for the design basis intact and damaged fuels. Since the source strengths of the damaged fuel and the MOX fuel are significantly smaller in all energy groups than those corresponding to the intact design basis fuel source strengths, the damaged and MOX fuel dose rates for normal conditions are bounded by the MPC-68 analysis with design basis intact fuel. Therefore, no explicit analysis is required to demonstrate that the MPC-68 with damaged or MOX fuel will meet the normal condition regulatory requirements.

Section 5.2.6 lists the gamma and neutron sources from the Dresden Unit 1 Thoria rod canister and demonstrates that the Thoria rod canister is bounded by the design basis Dresden Unit 1 6x6 intact fuel.

Section 5.2.4 presents the Co-60 sources from the BPRAs and TPDs that are permitted for storage in the HI-STAR 100. Section 5.4.6 demonstrates that the maximum dose rates presented in this section bound the dose rates from fuel assemblies containing either BPRAs or TPDs.

Section 5.4.7 demonstrates that the Dresden Unit 1 fuel assemblies containing antimony-beryllium neutron sources are bounded by the shielding analysis presented in this section.

Section 5.2.3 presents the gamma and neutron source for the design basis intact stainless steel clad fuel. The dose rates from this fuel are provided in Section 5.4.5.

The analyses summarized in this section demonstrate that the HI-STAR 100 System is in compliance with the 10CFR72.104 limits and ALARA practices.

5.1.2 Accident Conditions

The 10CFR72.106 radiation dose limits at the controlled area boundary for design basis accidents are:

> Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 Rem (0.05 Sv), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 Rem (0.5 Sv). The lens dose equivalent shall not exceed 15 Rem (0.15 Sv) and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem (0.5 Sv). The minimum distance from the spent fuel or high level radioactive waste handling and storage facilities to the nearest boundary of the controlled area shall be at least 100 meters.

The design basis accidents analyzed in Chapter 11 have one bounding consequence which affects the shielding materials. It is the damage to the neutron shield as a result of the design basis fire. Other design basis accidents result in damage to the outer enclosure shell and neutron shield; however, these accidents are localized. In a conservative fashion, the dose analysis assumes that as a result of the fire, the neutron shield is completely destroyed and replaced by a void. This is highly conservative as there will be limited sources of combustible materials stored in or around the ISFSI. Additionally, the neutron shield is assumed to be completely lost, whereas some portion of the neutron shield would be expected to remain, as the neutron shield material is fire retardant.

Throughout all design basis accident conditions the axial location of the fuel will remain fixed within the MPC because of the fuel spacers. Chapter 3 provides an analysis to show that the fuel spacers do not fail under all normal, off-normal, and accident conditions of storage. Chapter 3 also shows that the inner shell, intermediate shells, radial channels, and outer enclosure shell of the overpack remain unaltered throughout all design basis accident conditions. Localized damage of the overpack outer enclosure shell could be experienced. However, the localized deformations will have a negligible impact on the dose rate at the boundary of the controlled area.

The complete loss of the neutron shield significantly affects the dose at Dose Point #2 at the mid-height adjacent to the overpack neutron shield. Loss of the neutron shield has a small effect on the other dose points. To illustrate the impact of the design basis accident, the dose rates at Dose Point #2 (see Figure 5.1.1) are provided in Tables 5.1.8 and 5.1.9. The normal condition dose rates are provided for reference.

Table 5.1.9 provides a comparison of the normal and accident condition dose rates at one meter from the overpack. By comparing the increase in dose rates from normal and accident conditions and the maximum normal condition controlled area dose rate, it is evident that the dose as a result of the design basis accident cannot exceed 5 Rem (0.05 Sv) at the controlled area boundary for the short duration of the accident. Conservatively assuming a 1/R reduction in the dose rate, the dose rate at the 100 meter controlled area boundary would be less than 5 mrem/hr (0.05 mSv/h) for a single HI-STAR 100 during the accident condition. At this dose rate, it would take more than 1000 hours (41 days) for the dose at the controlled area boundary to reach 5 Rem (0.05 Sv). This length of time greatly exceeds the time necessary to implement and complete the corrective actions outlined in Chapter 11 for the fire accident. Therefore, the dose requirement of 10CFR72.106 is satisfied.

The consequences of the design basis accident conditions for the MPC-68 storing damaged fuel and the MPC-68F storing damaged fuel and/or fuel debris differ slightly from those with intact fuel. It is conservatively assumed that during a drop accident (vertical, horizontal, or tip-over) the fuel collapses and the pellets rest in the bottom of the damaged fuel container. Since the damaged and MOX fuels are both Dresden 1 fuel, the MOX fuel can also be considered damaged fuel. Analysis in Section 5.4.2 demonstrates that the damaged fuel in the post-accident condition has lower source terms (both gamma and neutron) per inch than the intact BWR design basis fuel. Therefore, the damaged fuel post-accident dose rates are bounded by the BWR intact fuel post-accident dose rates.

Analyses summarized in this section demonstrate the HI-STAR 100 System's compliance with the 10CFR72.106 limits.

Table 5.1.1

DOSE RATES ADJACENT TO OVERPACK FOR NORMAL CONDITIONS
MPC-32 WITH DESIGN BASIS ZIRCALOY CLAD FUEL AT WORST CASE
BURNUP AND COOLING TIME
40,000 MWD/MTU AND 8-YEAR COOLING
(The values in parentheses are in mSv/h)

Dose Point† Location	**Fuel Gammas†† (mrem/hr) (mSv/h)**	**^{60}Co Gammas (mrem/hr) (mSv/h)**	**Neutrons (mrem/hr) (mSv/h)**	**Totals (mrem/hr) (mSv/h)**
1	5.59 (0.0559)	220.07 (2.2007)	84.52 (0.8452)	310.18 (3.1018)
2	49.16 (0.4916)	0.03 (0.0003)	23.37 (0.2337)	72.56 (0.7256)
3	1.95 (0.0195)	80.91 (0.8091)	67.31 (0.6731)	150.16 (1.5016)
4	0.97 (0.0097)	35.84 (0.3584)	39.15 (0.3915)	75.96 (0.7596)
5	0.25 (0.0025)	0.58 (0.0058)	58.78 (0.5878)	59.6 (0.596)
6 (dry MPC)†††	12.96 (0.1296)	258.78 (2.5878)	141.89 (1.4189)	413.62 (4.1362)
7 (no temp. shield)	64.04 (0.6404)	1476.27 (14.7627)	438.45 (4.3845)	1978.75 (19.7875)
7 (with temp. shield)	20.4 (0.204)	323.69 (3.2369)	20.57 (0.2057)	364.66 (3.6466)

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

††† Overpack closure plate not present.

Table 5.1.2

DOSE RATES ADJACENT TO OVERPACK FOR NORMAL CONDITIONS
MPC-24 WITH DESIGN BASIS ZIRCALOY CLAD FUEL AT WORST CASE BURNUP AND COOLING TIME
40,000 MWD/MTU AND 5-YEAR COOLING
(The values in parentheses are in mSv/h)

Dose Point† Location	**Fuel Gammas†† (mrem/hr) (mSv/h)**	**^{60}Co Gammas (mrem/hr) (mSv/h)**	**Neutrons (mrem/hr) (mSv/h)**	**Totals (mrem/hr) (mSv/h)**
1	12.45 (0.1245)	231.52 (2.3152)	82.27 (0.8227)	326.24 (3.2624)
2	96.88 (0.9688)	0.03 (0.0003)	22.12 (0.2212)	119.03 (1.1903)
3	3.51 (0.0351)	81.12 (0.8112)	70.28 (0.7028)	154.90 (1.5490)
4	1.81 (0.0181)	35.86 (0.3586)	39.47 (0.3947)	77.14 (0.7714)
5	0.34 (0.0034)	0.69 (0.0069)	56.70 (0.5670)	57.73 (0.5773)
6 (dry MPC)†††	27.07 (0.2707)	286.19 (2.8619)	126.02 (1.2602)	439.28 (4.3928)
7 (no temp. shield)	100.36 (1.0036)	1432.28 (14.3228)	397.30 (3.9730)	1929.94 (19.2994)
7 (with temp. shield)	28.27 (0.2827)	329.84 (3.2984)	19.84 (0.1984)	377.94 (3.7794)

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

††† Overpack closure plate not present.

Table 5.1.3

DOSE RATES ADJACENT TO OVERPACK FOR NORMAL CONDITIONS
MPC-68 WITH DESIGN BASIS ZIRCALOY CLAD FUEL AT WORST CASE
BURNUP AND COOLING TIME
35,000 MWD/MTU AND 5-YEAR COOLING
(The values in parentheses are in mSv/h)

Dose Point† Location	Fuel Gammas†† (mrem/hr) (mSv/h)	^{60}Co Gammas (mrem/hr) (mSv/h)	Neutrons (mrem/hr) (mSv/h)	Totals (mrem/hr) (mSv/h)
1	10.26 (0.1026)	297.76 (2.9776)	65.63 (0.6563)	373.64 (3.7364)
2	100.42 (1.0042)	0.02 (0.0002)	19.40 (0.1940)	119.85 (1.1985)
3	0.97 (0.0097)	127.41 (1.2741)	29.88 (0.2988)	158.26 (1.5826)
4	0.44 (0.0044)	51.49 (0.5149)	17.76 (0.1776)	69.69 (0.6969)
5	0.13 (0.0013)	0.72 (0.0072)	26.45 (0.2645)	27.30 (0.2730)
6 (dry MPC)†††	9.32 (0.0932)	329.17 (3.2917)	65.38 (0.6538)	403.87 (4.0387)
7 (no temp. shield)	64.46 (0.6446)	1794.41 (17.9441)	325.90 (3.2590)	2184.76 (21.8476)
7 (with temp. shield)	20.36 (0.2036)	381.90 (3.8190)	14.52 (0.1452)	416.78 (4.1678)

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

††† Overpack closure plate not present.

Table 5.1.4

DOSE RATES AT ONE METER FOR NORMAL CONDITIONS
MPC-32 WITH DESIGN BASIS ZIRCALOY CLAD FUEL AT WORST CASE BURNUP AND COOLING TIME
40,000 MWD/MTU AND 8-YEAR COOLING
(The values in parentheses are in mSv/h)

Dose Point† Location	**Fuel Gammas†† (mrem/hr) (mSv/h)**	**^{60}Co Gammas (mrem/hr) (mSv/h)**	**Neutrons (mrem/hr) (mSv/h)**	**TOTALS (mrem/hr) (mSv/h)**
1	5.07 (0.0507)	23.84 (0.2384)	9.24 (0.0924)	38.14 (0.3814)
2	21.42 (0.2142)	0.63 (0.0063)	8.62 (0.0862)	30.67 (0.3067)
3	3.58 (0.0358)	13.42 (0.1342)	8.74 (0.0874)	25.75 (0.2575)
4	2.24 (0.0224)	14.56 (0.1456)	8.97 (0.0897)	25.78 (0.2578)
5	0.09 (0.0009)	0.28 (0.0028)	16.72 (0.1672)	17.1 (0.171)
7 (no temp. shield)	31.07 (0.3107)	712.91 (7.1291)	119.75 (1.1975)	863.73 (8.6373)
7 (with temp. shield)	7.27 (0.0727)	129.28 (1.2928)	15.14 (0.1514)	151.69 (1.5169)

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.5

DOSE RATES AT ONE METER FOR NORMAL CONDITIONS
MPC-24 WITH DESIGN BASIS ZIRCALOY CLAD FUEL AT WORST CASE
BURNUP AND COOLING TIME
40,000 MWD/MTU AND 5-YEAR COOLING
(The values in parentheses are in mSv/h)

Dose Point† Location	**Fuel Gammas†† (mrem/hr) (mSv/h)**	**^{60}Co Gammas (mrem/hr) (mSv/h)**	**Neutrons (mrem/hr) (mSv/h)**	**TOTALS (mrem/hr) (mSv/h)**
1	10.11 (0.1011)	25.00 (0.2500)	8.69 (0.0869)	43.79 (0.4379)
2	42.67 (0.4267)	1.06 (0.0106)	7.74 (0.0774)	51.47 (0.5147)
3	7.10 (0.0710)	14.06 (0.1406)	9.04 (0.0904)	30.21 (0.3021)
4	4.59 (0.0459)	14.69 (0.1469)	9.33 (0.0933)	28.61 (0.2861)
5	0.11 (0.0011)	0.32 (0.0032)	16.67 (0.1667)	17.11 (0.1711)
7 (no temp. shield)	52.66 (0.5266)	720.72 (7.2072)	116.30 (1.1630)	889.68 (8.8968)
7 (with temp. shield)	11.46 (0.1146)	139.22 (1.3922)	15.19 (0.1519)	165.87 (1.6587)

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.6

DOSE RATES AT ONE METER FOR NORMAL CONDITIONS
MPC-68 WITH DESIGN BASIS ZIRCALOY CLAD FUEL AT WORST CASE
BURNUP AND COOLING TIME
35,000 MWD/MTU AND 5-YEAR COOLING
(The values in parentheses are in mSv/h)

Dose Point† Location	**Fuel Gammas†† (mrem/hr) (mSv/h)**	**^{60}Co Gammas (mrem/hr) (mSv/h)**	**Neutrons (mrem/hr) (mSv/h)**	**Totals (mrem/hr) (mSv/h)**
1	10.47 (0.1047)	34.29 (0.3429)	7.52 (0.0752)	52.28 (0.5228)
2	43.01 (0.4301)	0.60 (0.0060)	7.50 (0.0750)	51.11 (0.5111)
3	4.37 (0.0437)	21.32 (0.2132)	4.23 (0.0423)	29.91 (0.2991)
4	2.60 (0.0260)	22.51 (0.2251)	4.19 (0.0419)	29.30 (0.2930)
5	0.06 (0.0006)	0.37 (0.0037)	7.44 (0.0744)	7.86 (0.0786)
7 (no temp. shield)	29.91 0.2991)	888.15 (8.8815)	87.01 (0.8701)	1005.07 (10.0507)
7 (with temp. shield)	8.10 (0.0810)	162.56 (1.6256)	10.52 (0.1052)	181.19 (1.8119)

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.7

DOSE RATES FOR ARRAYS
WITH DESIGN BASIS ZIRCALOY CLAD FUEL
(The values in parentheses are in mSv/year)

Array Configuration	1 cask	2x2	2x3	2x4	2x5
MPC-24 (40,000 MWD/MTU AND 5-YEAR COOLING)					
Annual Dose (mrem/year)† (mSv/year)	13.55 (0.1355)	18.60 (0.1860)	13.84 (0.1384)	18.45 (0.1845)	23.06 (0.2306)
Distance to Controlled Area Boundary (meters)††, †††	300	350	400	400	400
MPC-32 (40,000 MWD/MTU AND 8-YEAR COOLING)					
Annual Dose (mrem/year) † (mSv/year)	9.16 (0.0916)	13.96 (0.1396)	11.22 (0.1122)	14.96 (0.1496)	18.7 (0.1870)
Distance to Controlled Area Boundary (meters)††, †††	300	350	400	400	400

† 100% occupancy is assumed.

†† Dose location is at the center of the long side of the array.

††† Actual controlled area boundary dose rates will be lower because the maximum permissible burnup, as specified in Tables 2.1.13 and 2.1.15, for the stated cooling time is lower than the burnup analyzed for the design basis fuel used in this chapter.

Table 5.1.8

DOSE RATES ADJACENT TO OVERPACK FOR ACCIDENT CONDITIONS
DESIGN BASIS ZIRCALOY CLAD FUEL
AT WORST CASE BURNUP AND COOLING TIME
(The values in parentheses are in mSv/h)

Dose Point† Location	**Fuel Gammas†† (mrem/hr) (mSv/h)**	**^{60}Co Gammas (mrem/hr) (mSv/h)**	**Neutrons (mrem/hr) (mSv/h)**	**Totals (mrem/hr) (mSv/h)**
MPC-24 (40,000 MWD/MTU AND 5-YEAR COOLING)				
2 (Accident Condition)	221.84 (2.2184)	0.04 (0.0004)	1149.46 (11.4946)	1371.34 (13.7134)
2 (Normal Condition)	96.88 (0.9688)	0.03 (0.0003)	22.12 (0.2212)	119.03 (1.1903)
MPC-32 (40,000 MWD/MTU AND 8-YEAR COOLING)				
2 (Accident Condition)	106.86 (1.0686)	0.04 (0.0004)	1248.45 (12.4845)	1355.35 (13.5535)
2 (Normal Condition)	49.16 (0.4916)	0.03 (0.0003)	23.37 (0.2337)	72.56 (0.7256)
MPC-68 (35,000 MWD/MTU AND 5-YEAR COOLING)				
2 (Accident Condition)	223.20 (2.2320)	0.04 (0.0004)	1140.57 (11.4057)	1363.82 (13.6382)
2 (Normal Condition)	100.42 (1.0042)	0.02 (0.0002)	19.40 (0.1940)	119.85 (1.1985)

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.9

DOSE RATES AT ONE METER FOR ACCIDENT CONDITIONS
DESIGN BASIS ZIRCALOY CLAD FUEL
AT WORST CASE BURNUP AND COOLING TIME
(The values in parentheses provided are in mSv/h)

Dose Point† Location	**Fuel Gammas†† (mrem/hr) (mSv/h)**	**^{60}Co Gammas (mrem/hr) (mSv/h)**	**Neutrons (mrem/hr) (mSv/h)**	**Totals (mrem/hr) (mSv/h)**
MPC-24 (40,000 MWD/MTU AND 5-YEAR COOLING)				
2 (Accident Condition)	100.98 (1.0098)	1.80 (0.0180)	388.94 (3.8894)	491.73 (4.9173)
2 (Normal Condition)	42.67 (0.4267)	1.06 (0.0106)	7.74 (0.0774)	51.47 (0.5147)
MPC-32 (40,000 MWD/MTU AND 8-YEAR COOLING)				
2 (Accident Condition)	47.65 (0.4765)	1.6 (0.016)	421.85 (4.2185)	471.11 (4.7111)
2 (Normal Condition)	21.42 (0.2142)	0.63 (0.0063)	8.62 (0.0862)	30.67 (0.3067)
MPC-68 (35,000 MWD/MTU AND 5-YEAR COOLING)				
2 (Accident Condition)	98.28 (0.9828)	1.48 (0.0148)	360.93 (3.6093)	460.69 (4.6069)
2 (Normal Condition)	43.01 (0.4301)	0.60 (0.0060)	7.50 (0.0750)	51.11 (0.5111)

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

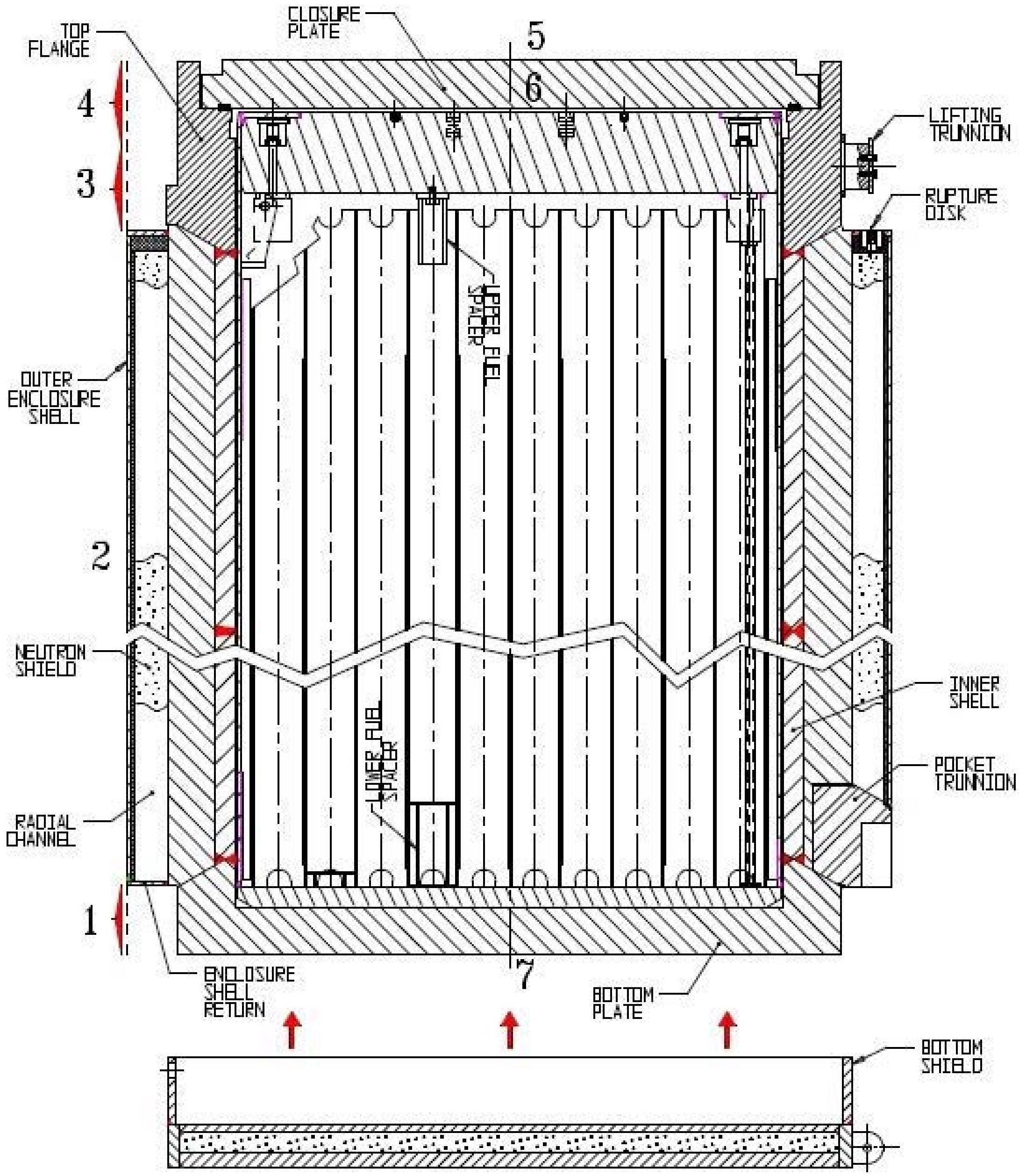


FIGURE 5.1.1; CROSS SECTION ELEVATION VIEW OF OVERPACK WITH DOSE POINT LOCATIONS

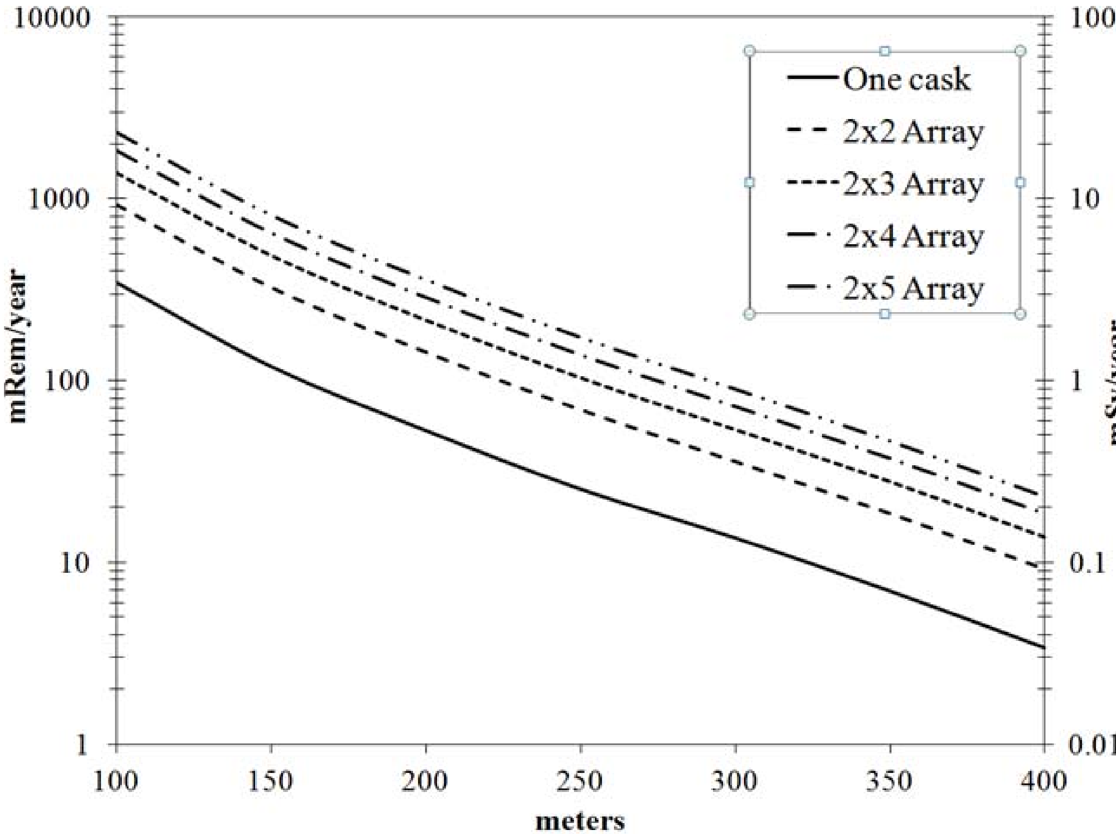


FIGURE 5.1.2; ANNUAL DOSE VERSUS FOR VARIOUS CONFIGURATIONS OF THE MPC-24 40,000 MWD/MTU AND 5-YEAR COOLING 100% OCCUPANCY ASSUMED

5.2 SOURCE SPECIFICATION

The neutron and gamma source terms, decay heat values, and quantities of radionuclides available for release, were calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.3 system [5.1.2, 5.1.3]. Sample input files for SAS2H and ORIGEN-S are provided in Appendices 5.A and 5.B, respectively. The gamma source term is actually comprised of three distinct sources. The first is a gamma source term from the active fuel region due to decay of fission products. The second source term is from ^{60}Co activity of the steel structural material in the fuel element above and below the active fuel region. The third source is from (n,γ) reactions described below.

A description of the design basis intact zircaloy clad fuel for the source term calculations is provided in Table 5.2.1. Multiple SAS2H and ORIGEN-S calculations were performed to confirm that the B&W 15x15 and the GE 7x7, which have the highest UO_2 mass, bound all other PWR and BWR fuel assemblies, respectively. Section 5.2.5 discusses, in detail, the determination of the design basis fuel assemblies.

The design basis Humboldt Bay and Dresden 1 6x6 fuel assembly, which is also the design basis damaged fuel assembly for the Humboldt Bay and Dresden 1 damaged fuel or fuel debris, is described in Table 5.2.2. The design basis damaged fuel assembly is also the design basis fuel assembly for fuel debris. The fuel assembly type listed produces the highest total neutron and gamma sources from the fuel assemblies at Dresden 1 and Humboldt Bay. Table 5.2.15 provides a description of the design basis Dresden 1 MOX fuel assembly used in this analysis. The design basis 6x6, damaged, and MOX fuel assemblies which are smaller than the GE 7x7, are assumed to have the same hardware characteristics as the GE 7x7. This is conservative because the larger hardware mass of the GE 7x7 results in a larger ^{60}Co activity.

The design basis stainless steel clad fuel assembly for the Haddam Neck and San Onofre 1 assembly classes is described in Table 5.2.18. This table also describes the design basis stainless steel clad LaCrosse fuel assembly.

In performing the SAS2H and ORIGEN-S calculations, a single full power cycle was used to achieve the desired burnup. This assumption, in conjunction with the above-average specific powers listed in Tables 5.2.1, 5.2.2, 5.2.15, and 5.2.18 resulted in conservative source term calculations.

Sections 5.2.1 and 5.2.2 describe the calculation of gamma and neutron source terms for zircaloy clad fuel while Section 5.2.3 discusses the calculation of the gamma and neutron source terms for the stainless steel clad fuel.

5.2.1 Gamma Source

Tables 5.2.3 through 5.2.6 provide the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design bases intact fuels for the MPC-32, MPC-24 and MPC-68, and the design basis damaged fuel. Table 5.2.16 provides the gamma source in MeV/s and photons/s for the design basis MOX fuel. NUREG-1536 [5.2.1] states that "only gammas with energies from approximately 0.8 to 2.5 MeV will contribute significantly to the dose rate." Conservatively, only energies in the range of 0.7 MeV-3.0 MeV are used in the shielding calculations. Photons with energies below 0.7 MeV are too weak to penetrate the steel of the overpack, and photons with energies above 3.0 MeV are too few to contribute significantly to the external dose. This section provides the radiation source for each of the burnup levels and cooling times evaluated.

The primary source of activity in the non-fuel regions of an assembly arise from the activation of ^{59}Co to ^{60}Co. The primary source of ^{59}Co in a fuel assembly is the steel structural material above and below the fuel. The zircaloy in these regions is neglected since it does not have a significant ^{59}Co impurity level. Reference [5.2.2] indicates that the impurity level in steel is 800 ppm or 0.8 gm/kg. As a conservative measure, the impurity level of ^{59}Co was assumed to be 1000 ppm or 1.0 gm/kg. Therefore, Inconel and stainless steel in the non-fuel regions are both conservatively assumed to have the 1.0 gm/kg impurity level.

The gamma source from the activation of the grid spacers is negligible in comparison to the source from the active fuel. In addition, in most fuel elements that obtain high burnups, the grid spacers are manufactured from zircaloy which does not activate to produce a gamma source. Therefore, for the PWR design basis fuel assembly, no contribution to the fuel region gamma source from activation of grid spacers is provided in the source term calculations. The BWR assembly grid spacers are zircaloy, however, some assembly designs have steel springs in conjunction with the grid spacers. The gamma source for the BWR fuel assembly includes the activation of these springs associated with the grid spacers.

The PWR fuel assemblies loaded into the MPC-32 basket may have non-zircaloy grid spacers. The calculations for PWR fuel assemblies with non-zircaloy grid spacers are provided in Subsection 5.4.9.

The non-fuel data listed in Table 5.2.1 was taken from References [5.2.2], [5.2.4], and [5.2.5]. The BWR masses are for an 8x8 fuel assembly. These masses are also appropriate for the 7x7 assembly since the masses of the non-fuel hardware from a 7x7 and an 8x8 are approximately the same. The masses listed are those of the steel components. The zircaloy in these regions was not included because zircaloy does not produce significant activation. These masses are larger than most other fuel assemblies from other manufactures. This, in combination with the conservative ^{59}Co impurity level, results in a conservative estimate of the ^{60}Co activity.

The masses in Table 5.2.1 were used to calculate a ^{59}Co impurity level in the fuel material. The grams of impurity were then used in ORIGEN-S to calculate a ^{60}Co activity level for the desired burnup and decay time. The methodology used to determine the activation level was developed from Reference [5.2.3] and is described here.

1. The activity of the ^{60}Co is calculated using ORIGEN-S. The flux used in the calculation was the in-core fuel region flux at full power.

2. The activity calculated in Step 1 for the region of interest was modified by the appropriate scaling factors listed in Table 5.2.7. These scaling factors were taken from Reference [5.2.3].

Tables 5.2.8 through 5.2.10 provide the ^{60}Co activity utilized in the shielding calculations in the non-fuel regions of the assemblies for the MPC-32, MPC-24 and the MPC-68. The design basis damaged and MOX fuel assemblies are conservatively assumed to have the same ^{60}Co source strength as the BWR intact design basis fuel. This is a conservative assumption as the design basis damaged fuel and MOX fuel are limited to a significantly lower burnup and longer cooling time than the intact design basis fuel.

In addition to the two sources already mentioned, a third source arises from (n,γ) reactions in the material of the MPC and the overpack. This source of photons is properly accounted for in MCNP when a neutron calculation is performed in a coupled neutron-gamma mode.

5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in order to obtain conservative source terms, low initial fuel enrichments were chosen for the BWR and PWR design basis fuel assemblies. The enrichments are appropriately varied as a function of burnup. Table 5.2.23 presents the ^{235}U initial enrichments for various burnup ranges from 20,000 - 50,000 MWD/MTU for PWR and BWR zircaloy clad fuel. These enrichments are based on Reference [5.2.6]. Table 8 of this reference presents average enrichments for burnup ranges. The initial enrichments chosen in Table 5.2.23 are approximately the average enrichments for the burnup range that are 5,000 MWD/MTU less than the ranges listed in Table 5.2.23. These enrichments are below the enrichments typically required to achieve the burnups that were analyzed. Therefore, the source term calculations are conservative.

The neutron source calculated for the design basis intact fuel assemblies for the MPC-32, MPC-24 and MPC-68 and the design basis damaged fuel are listed in Tables 5.2.11 through 5.2.14 in neutrons/s. Table 5.2.17 provides the neutron source in neutrons/sec for the design basis MOX fuel assembly. ^{244}Cm accounts for approximately 96% of the total number of neutrons produced, with slightly over 2% originating from (α,n) reactions within the UO_2 fuel. The remaining 2% derive from spontaneous fission in various Pu and Cm radionuclides. In addition, any neutrons generated from subcritical multiplication, (n,2n) or similar reactions are properly accounted for in the MCNP calculation.

5.2.3 Stainless Steel Clad Fuel Source

Table 5.2.18 lists the characteristics of the design basis stainless steel clad fuel. The fuel characteristics listed in this table are the input parameters that were used in the shielding calculations described in this chapter. The active fuel length listed in Table 5.2.18 is actually longer than the true active fuel length of 122 inches (310 cm) for the WE 15x15 and 83 inches (211cm) for the A/C 10x10. Since the true active fuel length is shorter than the design basis zircaloy clad active fuel length, it would be incorrect to calculate source terms for the stainless steel fuel using the correct fuel length and compare them directly to the zircaloy clad fuel source terms because this type of approach would not reflect the potential change in dose rates at the center of the cask (center of the active fuel). As an example, if it is assumed that the source strength for both the stainless steel and zircaloy fuel is 144 photons/s and that the active fuel lengths of the stainless steel fuel and zircaloy fuel are 83 inches (211 cm) and 144 inches (366 cm), respectively; the source strengths per inch of active fuel would be different for the two fuel types, 1.73 photons/s/inch (0.68 photons/s/cm) and 1 photons/s/inch (0.39 photons/s/cm) for the stainless steel and zircaloy fuel, respectively. The result would be a higher photon dose rate at the center of the cask with the stainless steel fuel than with the zircaloy clad fuel; a conclusion that would be overlooked by just comparing the source terms. This is an important consideration because the stainless steel clad fuel differs from the zircaloy clad in one important aspect: the stainless steel cladding will contain a significant photon source from Cobalt-60 which will be absent from the zircaloy clad fuel.

In order to eliminate the potential confusion when comparing source terms, the stainless steel clad fuel source terms were calculated with the same active fuel length as the design basis zircaloy clad fuel. Reference [5.2.2] indicates that the Cobalt-59 impurity level in steel is 800 ppm or 0.8 gm/kg. This impurity level was used for the stainless steel cladding in the source term calculations. It is assumed that the end fitting masses of the stainless steel clad fuel are the same as the end fitting masses of the zircaloy clad fuel. Therefore, separate source terms are not provided for the end fittings of the stainless steel fuel.

Tables 5.2.19 through 5.2.22 list the neutron and gamma source strengths for the design basis stainless steel clad fuel. It is obvious from these source terms that the neutron source strength for the stainless steel fuel is lower than for the zircaloy fuel. However, this is not true for all photon energy groups. The peak energy group is from 1.0 to 1.5 MeV which results from the large Cobalt activation in the cladding. Since some of the source strengths are higher for the stainless steel fuel, Section 5.4.5 presents the dose rates at the center of the overpack for the stainless steel fuel. The center dose location is the only location of concern since the end fittings are assumed to be the same mass as the end fittings for the zircaloy clad fuel. In addition, the burnup is lower and the cooling time is longer for the stainless steel fuel compared to the zircaloy clad fuel.

It should be noted that currently PWR spent fuel assemblies with stainless steel clad have not been qualified for storage in the MPC-32 basket.

5.2.4 Non-Fuel Hardware

Rod cluster control assemblies and axial power shaping rods are not permitted for storage in the HI-STAR 100 system. However, burnable poison rod assemblies (BPRAs) and thimble plug devices (TPDs) are permitted for storage in the HI-STAR 100 System as an integral part of a PWR fuel assembly.

5.2.4.1 BPRAs and TPDs

Burnable poison rod assemblies (BPRA) (including wet annular burnable absorbers and similarly designed devices with different names) and thimble plug devices (TPD) (including orifice rod assemblies, guide tube plugs, and similarly designed devices with different names) are an integral, yet removable, part of a large portion of PWR fuel. The TPDs are not used in all assemblies in a reactor core but are reused from cycle to cycle. Therefore, these devices can achieve very high burnups. In contrast, BPRAs are burned with a fuel assembly in core and are not reused. In fact, many BPRAs are removed after one or two cycles before the fuel assembly is discharged. Therefore, the achieved burnup for BPRAs is not significantly different than fuel assemblies.

TPDs are made of stainless steel and contain a small amount of inconel. These devices extend down into the plenum region of the fuel assembly but do not extend into the active fuel region with the exception of the W 14x14 water displacement guide tube plugs. Since these devices are made of stainless steel, there is a significant amount of cobalt-60 produced during irradiation. This is the only significant radiation source from the activation of steel and inconel.

BPRAs are made of stainless steel in the region above the active fuel zone and may contain a small amount of inconel in this region. Within the active fuel zone the BPRAs may contain 2-24 rodlets which are burnable absorbers clad in either zircaloy or stainless steel. The stainless steel clad BPRAs create a significant radiation source (Co-60) while the zircaloy clad BPRAs create a negligible radiation source. Therefore the stainless steel clad BPRAs are bounding.

SAS2H and ORIGEN-S were used to calculate a radiation source term and decay heat level for the TPDs and BPRAs. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating the appropriate mass of steel and inconel using the flux calculated for the design basis B&W 15x15 fuel assembly. The mass of material in the regions above the active fuel zone was scaled by the appropriate scaling factors listed in Table 5.2.10 in order to account for the reduced flux levels above the fuel assembly. The total curies of cobalt and the decay heat load were calculated for the TPDs and BPRAs as a function of burnup and cooling time. For burnups beyond 45,000 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 45,000 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 45,000 MWD/MTU.

Since the HI-STAR 100 cask system is designed to store many varieties of PWR fuel, a bounding TPD and BPRA had to be determined for the purposes of the analysis. This was accomplished by analyzing all of the BPRAs and TPDs (Westinghouse and B&W 14x14 through 17x17) found in references [5.2.5] and [5.2.7] to determine the TPD and BPRA which produced the highest Cobalt-60 source term and decay heat for a specific burnup and cooling time. The bounding TPD was determined to be the Westinghouse 17x17 guide tube plug and the bounding BPRA was actually determined by combining the higher masses of the Westinghouse 17x17 and 15x15 BPRAs into a singly hypothetical BPRA. The masses of this TPD and BPRA are listed in Table 5.2.29. As mentioned above, reference [5.2.5] describes the Westinghouse 14x14 water displacement guide tube plug as having a steel portion which extends into the active fuel zone. This particular water displacement guide tube plug was analyzed and determined to be bounded by the design basis TPD and BPRA.

Once the bounding BPRA and TPD were determined, the allowable decay heat load and Co-60 source from the BPRA and TPD were specified: 0.77 watts and 50 curies (1.85 TBq) Co-60 for each TPD, and 13.0 watts and 831 curies (30.747 TBq) Co-60 for each BPRA. Table 5.2.30 shows the curies of Co-60 that were calculated for BPRAs and TPDs in each region of the fuel assembly (e.g. incore, plenum, top). The allowable decay heat load for the TPDs and BPRAs was subtracted from the allowable decay heat load per assembly to determine the allowable PWR fuel assembly burnup and cooling times listed in Table 2.1.16. Since the decay heat load of the TPDs is negligible the same burnup and cooling time is used for assemblies with or without TPDs. However, a different burnup and cooling time is used for assemblies that contain BPRAs to account for the allowable BPRA decay heat load of 13.0 watts. A separate allowable burnup and cooling time is used for BPRAs and TPDs. These burnup and cooling times assure that the decay heat load and Cobalt-60 activity remain below the allowable levels specified above. It should be noted that at very high burnups, greater than 200,000 MWD/MTU the TPD decay heat load for a given cooling time actually decreases as the burnup continues to increase. This is due to a decrease in the Cobalt-60 production rate as the initial Cobalt-59 impurity is being depleted. Conservatively, a constant cooling time has been specified for burnups from 180,000 to 630,000 MWD/MTU for the TPDs.

Section 5.4.6 demonstrates that the dose rates from fuel assemblies containing BPRAs or TPDs are bounded by the dose rates presented in Section 5.1.1.

It should be noted that currently PWR spent fuel assemblies with BPRAs or TPDs have not been qualified for storage in the MPC-32 basket.

5.2.5 Choice of Design Basis Assembly

In order to perform a bounding analysis, a design basis fuel assembly must be chosen. Therefore, a fuel assembly from each fuel class was analyzed and a comparison of the neutrons/sec, photons/sec, and thermal power (watts) was performed. The fuel assembly which produced the highest source for a specified burnup, cooling time, and enrichment was chosen as the design basis fuel assembly. A separate design basis assembly was chosen for the MPC-24, MPC-32 and the MPC-68.

5.2.5.1 PWR Design Basis Assembly

Table 5.2.24 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad PWR fuel assembly. The fuel assembly listed for each class is the assembly with the highest UO_2 mass. The St. Lucie and Ft. Calhoun classes are not present in Table 5.2.24. These assemblies are shorter versions of the CE 16x16 and CE 14x14 assembly classes, respectively. Therefore, these assemblies are bounded by the CE 16x16 and CE 14x14 classes and were not explicitly analyzed. Since the Haddam Neck and San Onofre 1 classes are stainless steel clad fuel, these classes were analyzed separately and are discussed below. All fuel assemblies in Table 5.2.24 were analyzed at the same burnup and cooling time. The initial enrichment used in the analysis is consistent with Table 5.2.23. The results of the comparison are provided in Table 5.2.26. These results indicate that the B&W 15x15 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 5.2.24. This fuel assembly also has the highest UO_2 mass (see Table 5.2.24) which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO_2 mass produces the highest radiation source term.

The Haddam Neck and San Onofre 1 classes are shorter stainless steel clad versions of the WE 15x15 and WE 14x14 classes, respectively. Since these assemblies have stainless steel clad, they were analyzed separately as discussed in Section 5.2.3. Based on the results in Table 5.2.26, which show that the WE 15x15 assembly class has a higher source term than the WE 14x14 assembly class, the Haddam Neck, WE 15x15, fuel assembly was analyzed as the bounding PWR stainless steel clad fuel assembly.

5.2.5.2 BWR Design Basis Assembly

Table 5.2.25 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad BWR fuel assembly. The fuel assembly listed for each array type is the assembly that has the highest UO_2 mass. All fuel assemblies in Table 5.2.25 were analyzed at the same burnup and cooling time. The initial enrichment used in these analyses is consistent with Table 5.2.23. The results of the comparison are provided in Table 5.2.27. These results indicate that the 7x7 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 5.2.25. This fuel assembly also has the highest UO_2 mass which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO_2 mass produces the highest radiation source term. According to Reference [5.2.6], the last discharge of a 7x7 assembly was in 1985 and the maximum average burnup for a 7x7 during their operation was 29,000 MWD/MTU. This clearly indicates that the existing 7x7 assemblies have an average burnup and minimum cooling time that is well within the burnup and cooling time limits in Section 2.1. Therefore, the 7x7 assembly has never reached the burnup level analyzed in this chapter. However, in the interest of conservatism the 7x7 was chosen as the bounding fuel assembly array type.

Since the LaCrosse fuel assembly type is a stainless steel clad 10x10 assembly it was analyzed separately. The maximum burnup and minimum cooling times for this assembly are limited to 22,500 MWD/MTU and 10-year cooling as specified in Table 2.1.11. This assembly type is discussed further in Section 5.2.3.

The Humboldt Bay 6x6 and Dresden 1 6x6 fuel are older and shorter fuel than the other array types analyzed and therefore are considered separately. The Dresden 1 6x6 was chosen as the design basis fuel assembly for the Humboldt Bay 6x6 and Dresden 1 6x6 fuel assembly classes because it has the higher UO_2 mass. Dresden 1 also contains a few 6x6 MOX fuel assemblies which were explicitly analyzed as well.

Reference [5.2.6] indicates that the Dresden 1 6x6 fuel assembly has a higher UO_2 mass than the Dresden 1 8x8 or the Humboldt Bay fuel (6x6 and 7x7). Therefore, the Dresden 1 6x6 fuel assembly was also chosen as the bounding assembly for damaged fuel and fuel debris for the Humboldt Bay and Dresden 1 fuel assembly classes.

Since the design basis damaged fuel assembly and the design basis intact 6x6 fuel assembly are identical, the analysis presented in Section 5.4.2 for the damaged fuel assembly also demonstrates the acceptability of storing intact 6x6 fuel assemblies from the Dresden 1 and Humboldt Bay fuel assembly classes.

5.2.5.3 Decay Heat Loads

Section 2.1.5 describes the calculation of the burnup versus cooling time which is based on a maximum permissible decay heat per assembly. The decay heat values per assembly were calculated using the methodology described in Section 5.2. The design basis fuel assemblies, as described in Table 5.2.1, were used in the calculation of the burnup versus cooling time in Section 2.1. The enrichments used in the calculation of the decay heats were consistent with Table 5.2.23. As demonstrated in Tables 5.2.26 and 5.2.27, the design basis fuel assembly produces a higher decay heat value than the other assembly types considered. This is due to the higher heavy metal mass in the design basis fuel assemblies. Conservatively, Section 2.1 limits the heavy metal mass of the design basis fuel assembly classes to a value less than the design basis value utilized in this chapter. This provides additional assurance that the decay heat values are bounding values.

As further demonstration that the decay heat values (calculated using the design basis fuel assemblies) are conservative, a comparison between these calculated decay heats and the decay heats reported in Reference [5.2.7] are presented in Table 5.2.28. This comparison is made for a burnup of 30,000 MWD/MTU and a cooling time of 5 years. The burnup was chosen based on the limited burnup data available in Reference [5.2.7].

The heavy metal mass of the non-design basis fuel assembly classes in Section 2.1 are limited to the masses used in Tables 5.2.24 and 5.2.25. No margin is applied between the allowable mass and the analyzed mass of heavy metal for the non-design basis fuel assemblies. This is acceptable because additional assurance that the decay heat values for the non-design basis fuel assemblies are bounding values is obtained by using the decay heat values for the design basis fuel assemblies to determine the acceptable storage criteria for all fuel assemblies. As mentioned above, Table 5.2.28 demonstrates the level of conservatism in applying the decay heat from the design basis fuel assembly to all fuel assemblies.

5.2.6 Thoria Rod Canister

Dresden Unit 1 has a single DFC containing 18 thoria rods which have obtained a relatively low burnup, 16,000 MWD/MTU. These rods were removed from two 8x8 fuel assemblies which contained 9 rods each. The irradiation of thorium produces an isotope which is not commonly found in depleted uranium fuel. Th-232 when irradiated produces U-233. The U-233 can undergo an (n,2n) reaction which produces U-232. The U-232 decays to produce Tl-208 which produces a 2.6 MeV gamma during Beta decay. This results in a significant source in the 2.5-3.0 MeV range which is not commonly present in depleted uranium fuel. Therefore, this single DFC container was analyzed to determine if it was bounded by the current shielding analysis.

A radiation source term was calculated for the 18 thoria rods using SAS2H and ORIGEN-S for a burnup of 16,000 MWD/MTU and a cooling time of 18 years. Table 5.2.31 describes the 8x8 fuel assembly that contains the thoria rods. Table 5.2.32 and 5.2.33 show the gamma and neutron source terms, respectively, that were calculated for the 18 thoria rods in the thoria rod canister. Comparing these source terms to the design basis 6x6 source terms for Dresden Unit 1 fuel in Tables 5.2.6 and 5.2.14 clearly indicates that the design basis source terms bound the thoria rods source terms in all neutron groups and in all gamma groups except the 2.5-3.0 MeV group. As mentioned above, the thoria rods have a significant source in this energy range due to the decay of Tl-208.

Section 5.4.8 provides a further discussion of the thoria rod canister and its acceptability for storage in the HI-STAR 100 System.

5.2.7 Fuel Assembly Neutron Sources

Neutron sources are used in reactors during initial startup of reactor cores. There a different types of neutron sources (e.g. californium, americium-beryllium, plutonium-beryllium, antimony-beryllium). These neutron sources are typically inserted into the water rod of a fuel assembly and are usually removable.

Dresden Unit 1 has a few antimony-beryllium neutron sources. These sources have been analyzed in Section 5.4.7 to demonstrate that they are acceptable for storage in the HI-STAR 100 System. Currently these are the only neutron source permitted for storage in the HI-STAR 100 System.

Table 5.2.1

DESCRIPTION OF DESIGN BASIS INTACT ZIRCALOY CLAD FUEL
(The values in parentheses are in cm)

	PWR	**BWR**
Assembly type/class	B&W 15×15	GE 7×7
Active fuel length (in.) (cm)	144 (365.76)	144 (365.76)
No. of fuel rods	208	49
Rod pitch (in.) (cm)	0.568 (1.4427)	0.738 (1.8745)
Cladding material	zircaloy-4	zircaloy-2
Rod diameter (in.) (cm)	0.428 (1.0871)	0.570 (1.4478)
Cladding thickness (in.) (cm)	0.0230 (5.842E-2)	0.0355 (9.017E-2)
Pellet diameter (in.) (cm)	0.3742 (0.9505)	0.488 (1.2395)
Pellet material	UO_2	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	3.4 and 3.6	2.9 and 3.2
Burnup (MWD/MTU)†	40,000 and 47,500 (MPC-24) 40,000 and 45,000 (MPC-32)	35,000 and 45,000 (MPC-68)
Cooling Time (years)†	5 and 8 (MPC-24) 8 and 11 (MPC-32)	5 and 9 (MPC-68)
Specific power (MW/MTU)	40	30
Weight of UO_2 (kg)††	562.029	225.177
Weight of U (kg)††	495.485	198.516

Notes:
1. The B&W 15x15 is the design basis assembly for the fuel assembly classes listed in Table 5.2.24.
2. The GE 7x7 is the design basis assembly for the fuel assembly classes listed in Table 5.2.25.

† Burnup and cooling time combinations conservatively bound the acceptable burnup and cooling times listed in Section 2.1.

†† Derived from parameters in this table.

Table 5.2.1 (continued)

DESCRIPTION OF DESIGN BASIS INTACT ZIRCALOY CLAD FUEL
(The values in parentheses are in cm)

	PWR	**BWR**
No. of Water Rods/Guide Tubes	17	0
Water Rod O.D. (in.) (cm)	0.53 (1.3462)	N/A
Water Rod Thickness (in.) (cm)	0.0160 (4.064E-2)	N/A
Lower End Fitting (kg)	9.46	4.8
Gas Plenum Springs (kg)	0.72176	1.1
Gas Plenum Spacer (kg)	0.82824	N/A
Expansion Springs (kg)	N/A	0.4
Upper End Fitting (kg)	9.28	2.0
Handle (kg)	N/A	0.5
Fuel Grid Spacer Springs (kg of steel)	N/A	0.33

Table 5.2.2

DESCRIPTION OF DESIGN BASIS DAMAGED ZIRCALOY CLAD FUEL
(The values in parentheses are in cm)

	BWR
Fuel type	GE 6x6
Active fuel length (in.) (cm)	110 (279.4)
No. of fuel rods	36
Rod pitch (in.) (cm)	0.694 (1.7628)
Cladding material	zircaloy-2
Rod diameter (in.) (cm)	0.5645 (1.4338)
Cladding thickness (in.) (cm)	0.035 (8.89E-2)
Pellet diameter (in.) (cm)	0.494
Pellet material	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	2.24
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO_2 (kg)†	129.5
Weight of U (kg)†	114.2

Notes:
1. The 6x6 is the design basis damaged fuel assembly for the Humboldt Bay (all array types) and the Dresden 1 (all array types) damaged fuel assembly classes. It is also the design basis fuel assembly for the intact Humboldt Bay 6x6 and Dresden 1 6x6 fuel assembly classes.
2. This design basis damaged fuel assembly is also the design basis fuel assembly for fuel debris.

† Derived from parameters in this table.

Table 5.2.3

CALCULATED MPC-32 PWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	**Upper Energy**	**40,000 MWD/MTU 8-Year Cooling**		**45,000 MWD/MTU 11-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
7.0e-01	1.0	2.38E+14	2.80E+14	1.25E+14	1.48E+14
1.0	1.5	7.61E+13	6.09E+13	6.03E+13	4.82E+13
1.5	2.0	3.72E+12	2.13E+12	3.05E+12	1.74E+12
2.0	2.5	5.91E+11	2.63E+11	6.73E+10	2.99E+10
2.5	3.0	3.47E+10	1.26E+10	5.55E+09	2.02E+09
Totals		3.18E+14	3.43E+14	1.89E+14	1.98E+14

Table 5.2.4

CALCULATED MPC-24 PWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	**Upper Energy**	**40,000 MWD/MTU 5-Year Cooling**		**47,500 MWD/MTU 8-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
7.0e-01	1.0	5.96e+14	7.01e+14	3.06e+14	3.60e+14
1.0	1.5	1.38e+14	1.11e+14	9.68e+13	7.74e+13
1.5	2.0	8.94e+12	5.11e+12	4.61e+12	2.64e+12
2.0	2.5	6.85e+12	3.05e+12	6.28e+11	2.79e+11
2.5	3.0	2.67e+11	9.71e+10	3.96e+10	1.44e+10
Totals		7.50e+14	8.20e+14	4.08e+14	4.40e+14

Table 5.2.5

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	**Upper Energy**	**35,000 MWD/MTU 5-Year Cooling**		**45,000 MWD/MTU 9-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
7.0e-01	1.0	1.84e+14	2.17e+14	7.92e+13	9.32e+13
1.0	1.5	4.28e+13	3.43e+13	2.85e+13	2.28e+13
1.5	2.0	2.81e+12	1.60e+12	1.37e+12	7.83e+11
2.0	2.5	2.13e+12	9.48e+11	9.25e+10	4.11e+10
2.5	3.0	8.50e+10	3.09e+10	6.78e+9	2.47e+9
Totals		2.32e+14	2.54e+14	1.09e+14	1.17e+14

Table 5.2.6

CALCULATED MPC-68 and MPC-68F BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD DAMAGED FUEL

Lower Energy	**Upper Energy**	**30,000 MWD/MTU 18-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
7.0e-01	1.0	3.97e+12	4.67e+12
1.0	1.5	3.67e+12	2.94e+12
1.5	2.0	2.20e+11	1.26e+11
2.0	2.5	1.35e+9	5.99e+8
2.5	3.0	7.30e+7	2.66e+7
Totals		7.86e+12	7.74e+12

Table 5.2.7

SCALING FACTORS USED IN CALCULATING THE ^{60}Co SOURCE

Region	**PWR**	**BWR**
Handle	N/A	0.05
Upper end fitting	0.1	0.1
Gas plenum spacer	0.1	N/A
Expansion springs	N/A	0.1
Gas plenum springs	0.2	0.2
Grid spacer springs	N/A	1.0
Lower end fitting	0.2	0.15

Table 5.2.8

CALCULATED MPC-32 ^{60}Co SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
AT VARYING BURNUP AND COOLING TIMES
(The values in parentheses are in TBq)

Location	**40,000 MWD/MTU 8-Year Cooling (curies) (TBq)**	**45,000 MWD/MTU 11-Year Cooling (curies) (TBq)**
Lower end fitting	104.44 (3.864)	75.87 (2.807)
Gas plenum springs	7.97 (0.295)	5.79 (0.214)
Gas plenum spacer	4.57 (0.169)	3.32 (0.123)
Expansion springs	N/A	N/A
Grid spacer springs	N/A	N/A
Upper end fitting	51.23 (1.895)	37.21 (1.377)
Handle	N/A	N/A

Table 5.2.9

CALCULATED MPC-24 ^{60}Co SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
AT VARYING BURNUP AND COOLING TIMES
(The values in parentheses are in TBq)

Location	**40,000 MWD/MTU 5-Year Cooling (curies) (TBq)**	**47,500 MWD/MTU 8-Year Cooling (curies) (TBq)**
Lower end fitting	154.95 (5.7332)	118.06 (4.3682)
Gas plenum springs	11.82 (0.4373)	9.01 (0.3334)
Gas plenum spacer	6.78 (0.2509)	5.17 (0.1913)
Expansion springs	N/A	N/A
Grid spacer springs	N/A	N/A
Upper end fitting	76.00 (2.812)	57.91 (2.1427)
Handle	N/A	N/A

Table 5.2.10

CALCULATED MPC-68 ^{60}Co SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
AT VARYING BURNUP AND COOLING TIMES
(The values in parentheses are in TBq)

Location	**35,000 MWD/MTU 5-Year Cooling (curies) (TBq)**	**45,000 MWD/MTU 9-Year Cooling (curies) (TBq)**
Lower end fitting	57.38 (2.1231)	40.15 (1.4856)
Gas plenum springs	17.53 (0.6486)	12.27 (0.4540)
Gas plenum spacer	N/A	N/A
Expansion springs	3.19 (0.1180)	2.23 (0.0825)
Grid spacer springs	26.30 (0.9731)	18.40 (0.6808)
Upper end fitting	15.94 (0.5898)	11.15 (0.4126)
Handle	1.99 (0.0736)	1.39 (0.0514)

Table 5.2.11

CALCULATED MPC-32 PWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUP AND COOLING TIMES

Lower Energy (MeV)	**Upper Energy (MeV)**	**40,000 MWD/MTU 8-Year Cooling (Neutrons/s)**	**45,000 MWD/MTU 11-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	9.94E+06	1.30E+07
4.0e-01	9.0e-01	5.08E+07	6.63E+07
9.0e-01	1.4	4.65E+07	6.08E+07
1.4	1.85	3.44E+07	4.49E+07
1.85	3.0	6.10E+07	7.95E+07
3.0	6.43	5.51E+07	7.19E+07
6.43	20.0	4.87E+06	6.35E+06
Totals		2.63E+08	3.43E+08

Table 5.2.12

CALCULATED MPC-24 PWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUP AND COOLING TIMES

Lower Energy (MeV)	Upper Energy (MeV)	40,000 MWD/MTU 5-Year Cooling (Neutrons/s)	47,500 MWD/MTU 8-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	1.11e+7	1.80e+7
4.0e-01	9.0e-01	5.69e+7	9.19e+7
9.0e-01	1.4	5.21e+7	8.41e+7
1.4	1.85	3.85e+7	6.20e+7
1.85	3.0	6.80e+7	1.10e+8
3.0	6.43	6.16e+7	9.94e+7
6.43	20.0	5.45e+6	8.81e+6
Totals		2.94e+8	4.74e+8

Table 5.2.13

CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUP AND COOLING TIMES

Lower Energy (MeV)	**Upper Energy (MeV)**	**35,000 MWD/MTU 5-Year Cooling (Neutrons/s)**	**45,000 MWD/MTU 9-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	3.05e+6	6.23e+6
4.0e-01	9.0e-01	1.56e+7	3.18e+7
9.0e-01	1.4	1.43e+7	2.91e+7
1.4	1.85	1.06e+7	2.15e+7
1.85	3.0	1.87e+7	3.79e+7
3.0	6.43	1.69e+7	3.44e+7
6.43	20.0	1.49e+6	3.05e+6
Totals		8.06e+7	1.64e+8

Table 5.2.14

CALCULATED MPC-68 and MPC-68F BWR NEUTRON SOURCE PER ASSEMBLY FOR DESIGN BASIS DAMAGED ZIRCALOY CLAD FUEL

Lower Energy (MeV)	Upper Energy (MeV)	30,000 MWD/MTU 18-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	8.22e+5
4.0e-01	9.0e-01	4.20e+6
9.0e-01	1.4	3.87e+6
1.4	1.85	2.88e+6
1.85	3.0	5.18e+6
3.0	6.43	4.61e+6
6.43	20.0	4.02e+5
Totals		2.20e+7

Table 5.2.15

DESCRIPTION OF DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL
(The values in parentheses are in cm)

	BWR
Fuel type	GE 6x6
Active fuel length (in.) (cm)	110 (279.4)
No. of fuel rods	36
Rod pitch (in.) (cm)	0.696 1.7678)
Cladding material	zircaloy-2
Rod diameter (in.) (cm)	0.5645 (1.4338)
Cladding thickness (in.) (cm)	0.036 (9.144E-2)
Pellet diameter (in.) (cm)	0.482 (1.2243)
Pellet material	UO_2 and $PuUO_2$
No. of UO_2 Rods	27
No. of $PuUO_2$ rods	9
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)†	2.24 (UO_2 rods) 0.711 ($PuUO_2$ rods)
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO_2,$PuUO_2$ (kg)††	123.3
Weight of U,Pu (kg)††	108.7

† See Table 5.3.3 for detailed composition of $PuUO_2$ rods.

†† Derived from parameters in this table.

Table 5.2.16

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

Lower Energy	**Upper Energy**	**30,000 MWD/MTU 18-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
7.0e-01	1.0	3.87e+12	4.56e+12
1.0	1.5	3.72e+12	2.98e+12
1.5	2.0	2.18e+11	1.25e+11
2.0	2.5	1.17e+9	5.22e+8
2.5	3.0	9.25e+7	3.36e+7
Totals		7.81e+12	7.67e+12

Table 5.2.17

CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

Lower Energy (MeV)	**Upper Energy (MeV)**	**30,000 MWD/MTU 18-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	1.24e+6
4.0e-01	9.0e-01	6.36e+6
9.0e-01	1.4	5.88e+6
1.4	1.85	4.43e+6
1.85	3.0	8.12e+6
3.0	6.43	7.06e+6
6.43	20.0	6.07e+5
Totals		3.37e+7

Table 5.2.18

DESCRIPTION OF DESIGN BASIS INTACT STAINLESS STEEL CLAD FUEL
(The values in parentheses are in cm)

	PWR	BWR
Fuel type	WE 15x15	A/C 10x10
Active fuel length (in.) (cm)	144 (365.76)	144 (365.76)
No. of fuel rods	204	100
Rod pitch (in.) (cm)	0.563 (1.43)	0.565 (1.4351)
Cladding material	304 SS	348H SS
Rod diameter (in.) (cm)	0.422 (1.0719)	0.396 (1.0058)
Cladding thickness (in.) (cm)	0.0165 (4.191E-2)	0.02 (5.08E-2)
Pellet diameter (in.) (cm)	0.3825 (0.9716)	0.35 (0.889)
Pellet material	UO_2	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	3.5	3.5
Burnup (MWD/MTU)	30,000 @ 9 yr (MPC-24) 40,000 @ 15 yr (MPC-24)	22,500 (MPC-68)
Cooling Time (years)	9 (MPC-24) 15 (MPC-24)	10 (MPC-68)
Specific power (MW/MTU)	37.96	29.17
No. of Water Rods	21	0
Water Rod O.D. (in.) (cm)	0.546 (1.3868)	N/A
Water Rod Thickness (in.) (cm)	0.017 (4.318E-2)	N/A

Table 5.2.19

CALCULATED BWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy	**Upper Energy**	**22,500 MWD/MTU 10-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
7.0e-01	1.0	1.97e+13	2.31e+13
1.0	1.5	7.93e+13	6.34e+13
1.5	2.0	4.52e+11	2.58e+11
2.0	2.5	3.28e+10	1.46e+10
2.5	3.0	1.69e+9	6.14e+8
Totals		9.95e+13	8.68e+13

Note:
These source terms were calculated for a 144 inch (365.76 cm) active fuel length. The actual active fuel length is 83 inches (210.82 cm).

Table 5.2.20

CALCULATED PWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy	**Upper Energy**	**30,000 MWD/MTU 9-Year Cooling**		**40,000 MWD/MTU 15-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
7.0e-01	1.0	1.18e+14	1.39e+14	4.79e+13	5.63e+13
1.0	1.5	3.00e+14	2.40e+14	1.88e+14	1.50e+14
1.5	2.0	2.28e+12	1.30e+12	2.07e+12	1.18e+12
2.0	2.5	2.34e+11	1.04e+11	1.28e+10	5.71e+9
2.5	3.0	1.33e+10	4.83e+9	9.59e+8	3.49e+8
Totals		4.21e+14	3.80e+14	2.38e+14	2.07e+14

Note:
These source terms were calculated for a 144 inch (365.76 cm) active fuel length. The actual active fuel length is 122 inches (309.88 cm).

Table 5.2.21

CALCULATED BWR NEUTRON SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy (MeV)	**Upper Energy (MeV)**	**22,500 MWD/MTU 10-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	2.23e+5
4.0e-01	9.0e-01	1.14e+6
9.0e-01	1.4	1.07e+6
1.4	1.85	8.20e+5
1.85	3.0	1.56e+6
3.0	6.43	1.30e+6
6.43	20.0	1.08e+5
Total		6.22e+6

Note:
These source terms were calculated for a 144 inch (365.76 cm) active fuel length. The actual active fuel length is 83 inches (210.82 cm).

Table 5.2.22

CALCULATED PWR NEUTRON SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy (MeV)	**Upper Energy (MeV)**	**30,000 MWD/MTU 9-Year Cooling (Neutrons/s)**	**40,000 MWD/MTU 15-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	3.05e+6	8.02e+6
4.0e-01	9.0e-01	1.56e+7	4.10e+7
9.0e-01	1.4	1.44e+7	3.77e+7
1.4	1.85	1.07e+7	2.79e+7
1.85	3.0	1.93e+7	4.98e+7
3.0	6.43	1.71e+7	4.47e+7
6.43	20.0	1.49e+6	3.93e+6
Totals		8.16e+7	2.13e+8

Note:
These source terms were calculated for a 144 inch (365.76 cm) active fuel length. The actual active fuel length is 122 inches (309.88 cm).

Table 5.2.23

INITIAL ENRICHMENTS USED IN THE SOURCE TERM CALCULATIONS

Burnup Range (MWD/MTU)	Initial Enrichment (wt.% ^{235}U)
BWR Fuel	
20,000-25,000	2.1
25,000-30,000	2.4
30,000-35,000	2.6
35,000-40,000	2.9
40,000-45,000	3.0
45,000-50,000	3.2
PWR Fuel	
20,000-25,000	2.3
25,000-30,000	2.6
30,000-35,000	2.9
35,000-40,000	3.2
40,000-45,000	3.4
45,000-50,000	3.6

Note: The burnup ranges do not overlap. Therefore, 20,000-25,000 MWD/MTU means 20,000-24,999.9 MWD/MTU, etc.

Table 5.2.24

DESCRIPTION OF EVALUATED INTACT ZIRCALOY CLAD PWR FUEL
(The values in parentheses are in cm)

Assembly class	WE 14×14	WE 15×15	WE 17×17	CE 14×14	CE 16×16	B&W 15×15	B&W 17×17
Active fuel length (in.) (cm)	144 (365.76)	144 (365.76)	144 (365.76)	144 (365.76)	150 (365.76)	144 (365.76)	144 (365.76)
No. of fuel rods	179	204	264	176	236	208	264
Rod pitch (in.) (cm)	0.556 (1.4122)	0.563 (1.43)	0.496 (1.2598)	0.580 (1.4732)	0.5063 (1.286)	0.568 (1.4427)	0.502 (1.2751)
Cladding material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Rod diameter (in.) (cm)	0.422 (1.0719)	0.422 (1.0719)	0.374 (0.95)	0.440 (1.1176)	0.382 (0.9703)	0.428 (1.0871)	0.377 (0.9576)
Cladding thickness (in.) (cm)	0.0243 (0.0617)	0.0245 (0.0622)	0.0225 (0.0572)	0.0280 (0.0711)	0.0250 (0.0635)	0.0230 (0.0584)	0.0220 (0.0559)
Pellet diameter (in.) (cm)	0.3659 (0.9294)	0.366 (0.9296)	0.3225 (0.8192)	0.377 (0.9576)	0.3255 (0.8268)	0.3742 (0.9505)	0.3252 (0.8260)
Pellet material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Pellet density (gm/cc) (95% of theoretical)	10.412	10.412	10.412	10.412	10.412	10.412	10.412
Enrichment (wt.% ^{235}U)	3.4	3.4	3.4	3.4	3.4	3.4	3.4
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5	5	5	5
Specific power (MW/MTU)	40	40	40	40	40	40	40
Weight of UO_2 (kg)†	462.451	527.327	529.848	482.706	502.609	562.029	538.757
Weight of U (kg)†	407.697	464.891	467.114	425.554	443.100	495.485	474.968
No. of Guide Tubes	17	21	25	5	5	17	25
Guide Tube O.D. (in.) (cm)	0.539 (1.3691)	0.546 (1.3868)	0.474 (1.204)	1.115 (2.8321)	0.98 (2.4892)	0.53 (1.3462)	0.564 (1.4326)
Guide Tube Thickness (in.) (cm)	0.0170 (0.0432)	0.0170 (0.0432)	0.0160 (0.0406)	0.0400 (0.1016)	0.0400 (0.1016)	0.0160 (0.0406)	0.0175 (0.0445)

† Derived from parameters in this table.

Table 5.2.25

DESCRIPTION OF EVALUATED INTACT ZIRCALOY CLAD BWR FUEL
(The values in parentheses are in cm)

Array Type	7×7	8×8	9×9	10×10
Active fuel length (in.) (cm)	144 (365.76)	144 (365.76)	144 (365.76)	144 (365.76)
No. of fuel rods	49	63	74	92
Rod pitch (in.) (cm)	0.738 (1.8745)	0.640 (1.6526)	0.566 (1.4376)	0.510 (1.2954)
Cladding material	Zr-2	Zr-2	Zr-2	Zr-2
Rod diameter (in.) (cm)	0.570 (1.4478)	0.493 (1.2522)	0.440 (1.1176)	0.404 (1.0262)
Cladding thickness (in.) (cm)	0.0355 (0.0902)	0.0340 (0.0864)	0.0280 (0.0711)	0.0260 (0.0660)
Pellet diameter (in.) (cm)	0.488 (1.2395)	0.416 (1.0566)	0.376 (0.955)	0.345 (0.8763)
Pellet material	UO_2	UO_2	UO_2	UO_2
Pellet density (gm/cc) (95% of theoretical)	10.412	10.412	10.412	10.412
Enrichment (wt.% ^{235}U)	3.0	3.0	3.0	3.0
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5
Specific power (MW/MTU)	30	30	30	30
Weight of UO_2 (kg)†	225.177	210.385	201.881	211.307
Weight of U (kg)†	198.516	185.475	177.978	186.288
No. of Water Rods	0	1	2	2
Water Rod O.D. (in.) (cm)	n/a	0.493 (1.2522)	0.980 (2.4892)	0.980 (2.4892)
Water Rod Thickness (in.) (cm)	n/a	0.0340 (0.0864)	0.0300 (0.0762)	0.0300 (0.0762)

† Derived from parameters in this table.

Table 5.2.26

COMPARISON OF SOURCE TERMS FOR INTACT ZIRCALOY CLAD PWR FUEL
3.4 wt.% ^{235}U - 40,000 MWD/MTU - 5 years cooling

Assembly class	WE 14×14	WE 15×15	WE 17×17	CE 14×14	CE 16×16	B&W 15×15	B&W 17×17
Neutrons/sec	2.29e+8 / 2.31e+8	2.63e+8 / 2.65e+8	2.62e+8	2.31e+8	2.34e+8	2.94e+8	2.64e+8
Photons/sec (0.7-3.0 MeV)	6.64e+14/ 7.11e+14	7.54e+14/ 8.12e+14	7.60e+14	6.77e+14	7.06e+14	8.20e+14	7.71e+14
Thermal power (watts)	926.6 / 936.8	1056 / 1068	1062	956.6	995.7	1137	1077

Note:
The WE 14x14 and WE 15x15 have both zircaloy and stainless steel guide tubes. The first value presented is for the assembly with zircaloy guide tubes and the second value is for the assembly with stainless steel guide tubes.

Table 5.2.27

COMPARISON OF SOURCE TERMS FOR INTACT ZIRCALOY CLAD BWR FUEL
3.0 wt.% ^{235}U - 40,000 MWD/MTU - 5 years cooling

Assembly class	7×7	8×8	9×9	10×10
Neutrons/sec	1.33e+8	1.17e+8	1.11e+8	1.22e+8
Photons/sec (0.7-3.0 MeV)	3.10e+14	2.83e+14	2.71e+14	2.89e+14
Thermal power (watts)	435.5	402.3	385.3	407.4

Table 5.2.28

COMPARISON OF CALCULATED DECAY HEATS FOR DESIGN BASIS FUEL
AND VALUES REPORTED IN THE
DOE CHARACTERISTICS DATABASE † FOR
30,000 MWD/MTU AND 5-YEAR COOLING

Fuel Assembly Class	Decay Heat from the DOE Database (watts/assembly)	Decay Heat from Design Basis Fuel (watts/assembly)
PWR Fuel		
B&W 15x15	752.0	827.5
B&W 17x17	732.9	827.5
CE 16x16	653.7	827.5
CE 14x14	601.3	827.5
WE 17x17	742.5	827.5
WE 15x15	762.2	827.5
WE 14x14	649.6	827.5
BWR Fuel		
7x7	310.9	315.7
8x8	296.6	315.7
9x9	275.0	315.7

Notes:
1. The PWR and BWR design basis fuels are the B&W 15x15 and the GE 7x7, respectively.
2. The decay heat values from the database include contributions from in-core material (e.g. spacer grids).
3. Information on the 10x10 was not available in the DOE database. However, based on the results in Table 5.2.27, the actual decay heat values from the 10x10 would be very similar to the values shown above for the 8x8.

† Reference [5.2.7].

Table 5.2.29

DESCRIPTION OF DESIGN BASIS BURNABLE POISON ROD ASSEMBLY AND THIMBLE PLUG DEVICE

Region	**BPRA**	**TPD**
Upper End Fitting (kg of steel)	2.62	2.3
Upper End Fitting (kg of inconel)	0.42	0.42
Gas Plenum Spacer (kg of steel)	0.77488	1.71008
Gas Plenum Springs (kg of steel)	0.67512	1.48992
In-core (kg of steel)	13.2	N/A

Table 5.2.30

DESIGN BASIS COBALT-60 ACTIVITIES FOR BURNABLE POISON ROD ASSEMBLIES AND THIMBLE PLUG DEVICES
(The values in parentheses are in TBq)

Region	**BPRA**	**TPD**
Upper End Fitting (curies Co-60) (TBq Co-60)	30.4 (1.1248)	25.21 (0.9328)
Gas Plenum Spacer (curies Co-60) (TBq Co-60)	4.6 (0.1702)	9.04 (0.3345)
Gas Plenum Springs (curies Co-60) (TBq Co-60)	8.2 (0.3034)	15.75 (0.5828)
In-core (curies Co-60) (TBq Co-60)	787.8 (29.1486)	N/A

Table 5.2.31

DESCRIPTION OF FUEL ASSEMBLY
THORIA RODS IN THE THORIA ROD CANISTER
(The values in parentheses are in cm)

	BWR
Fuel type	8x8
Active fuel length (in.) (cm)	110.5 (280.67)
No. of UO_2 fuel rods	55
No. of UO_2/ThO_2 fuel rods	9
Rod pitch (in.) (cm)	0.523 (1.3284)
Cladding material	zircaloy
Rod diameter (in.) (cm)	0.412 (1.0465)
Cladding thickness (in.) (cm)	0.025 (0.0635)
Pellet diameter (in.) (cm)	0.358 (0.9093)
Pellet material	98.5% ThO_2 and 1.5% UO_2 for UO_2/ThO_2 rods
Pellet density (gm/cc)	10.412
Enrichment (w/o ^{235}U)	93.5 in UO_2 for UO_2/ThO_2 rods and 1.8 for UO_2 rods
Burnup (MWD/MTIHM)	16,000
Cooling Time (years)	18
Specific power (MW/MTIHM)	16.5
Weight of THO_2 and UO_2 (kg)†	121.46
Weight of U (kg)†	92.29
Weight of Th (kg)†	14.74

† Derived from parameters in this table.

Table 5.2.32

CALCULATED FUEL GAMMA SOURCE FOR THORIA ROD CANISTER CONTAINING EIGHTEEN THORIA RODS

Lower Energy	**Upper Energy**	**16,000 MWD/MTIHM 18-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
7.0e-01	1.0	5.79e+11	6.81e+11
1.0	1.5	3.79e+11	3.03e+11
1.5	2.0	4.25e+10	2.43e+10
2.0	2.5	4.16e+8	1.85e+8
2.5	3.0	2.31e+11	8.39e+10
Totals		1.23e+12	1.09e+12

The pellet material of Thoria rod used for the above source term is slightly different from that provided in Table 5.2.31. Since the difference is a small fraction of the allowable value (i.e., 0.3%), it is reasonable to conclude that the increase in Thoria will have a negligible effect on the resultant neutron spectra.

Table 5.2.33

CALCULATED FUEL NEUTRON SOURCE FOR THORIA ROD CANISTER CONTAINING EIGHTEEN THORIA RODS

Lower Energy (MeV)	**Upper Energy (MeV)**	**16,000 MWD/MTIHM 18-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	5.65e+2
4.0e-01	9.0e-01	3.19e+3
9.0e-01	1.4	6.79e+3
1.4	1.85	1.05e+4
1.85	3.0	3.68e+4
3.0	6.43	1.41e+4
6.43	20.0	1.60e+2
Totals		7.21e+4

The pellet material of Thoria rod used to extract the above source term is slightly different from that provided in Table 5.2.31. Since the difference is a small fraction of the allowable value (i.e., 0.3%), it is reasonable to conclude that the increase in Thoria will have a negligible effect on the resultant gamma spectra.

5.3 MODEL SPECIFICATIONS

The shielding analysis of the HI-STAR 100 System was performed with MCNP-4A [5.1.1]. MCNP is a Monte Carlo transport code that offers a full three-dimensional combinatorial geometry modeling capability including such complex surfaces as cones and tori. This means that no gross approximations were required to represent the HI-STAR 100 System in the shielding analysis. A sample input file for MCNP is provided in Appendix 5.C.

As discussed in Section 5.1.1, off-normal conditions do not have any implications for the shielding analysis. Therefore, the MCNP models and results developed for the normal conditions also represent the off-normal condition. Section 5.1.2 discussed the accident conditions and stated that the only accident that would impact the shielding analysis would be a loss of the neutron shield. Therefore, the MCNP models of the HI-STAR 100 System normal condition have the neutron shield in place while the accident condition replaces the neutron shield with void.

5.3.1 Description of the Radial and Axial Shielding Configuration

Section 1.5 provides the drawings that describe the HI-STAR 100 System. These drawings were used to create the MCNP models used in the radiation transport calculations. Figures 5.3.1 through 5.3.3 show cross sectional views of the HI-STAR 100 overpack and MPC as it was modeled in MCNP for each of the MPCs. These figures were created with the MCNP two-dimensional plotter and are drawn to scale. The figures clearly illustrate the radial steel fins and pocket trunnions in the neutron shield region. Since the fins and pocket trunnions were modeled explicitly, neutron streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and 1 meter dose. In Section 5.4.1, the dose effect of localized streaming through these compartments is analyzed. Figure 5.3.4 shows the MCNP models of the MPC-32 fuel basket. Figures 5.3.5 and 5.3.6 show the MCNP models of the MPC-24 and MPC-68 fuel baskets including the as-modeled dimensions. Figure 5.3.9 shows a cross sectional view of the HI-STAR 100 overpack with the as-modeled thickness of the various materials. Figure 5.3.10 is an axial representation of the HI-STAR 100 overpack with the various as-modeled dimensions indicated. It should be noted that the thickness of the MPC-68 lid is 10.0 inches (25.4 cm), and the thickness of the MPC-24 lid and MPC-32 lid is 9.5 inches (24.13 cm). Correspondingly, the MPC-internal cavity height of MPC-68 differs by 0.5 inch (1.27 cm) compared to that of MPC-24 and MPC-32. In the MCNP models of the MPC-24, MPC-32 and MPC-68, the actual lid thickness and internal cavity height for that particular MPC was used.

Calculations were performed to determine the acceptability of homogenizing the fuel assembly versus explicit modeling. Based on these calculations it was concluded that it was acceptable to homogenize the fuel assembly without loss of accuracy. The PWR fuel assembly modeled was the design basis fuel assembly, the B&W 15x15. The width of this homogenized fuel assembly in MCNP is equal to 15 times the pitch. The BWR fuel assembly modeled was an 8x8 fuel assembly. This is different from the 7x7 design basis fuel assembly used for the source term calculations. However, it is conservative to use an 8x8 fuel assembly in the MCNP model since it contains less fuel and therefore less shielding than the 7x7 fuel assembly. The width of the BWR homogenized fuel assembly is equal to 8 times the pitch. Homogenization of the fuel assemblies resulted in a noticeable decrease in run time.

Several conservative approximations were made in modeling the MPC. The conservative approximations are listed below.

1. The basket material in the top and bottom 0.9 inches (2.286 cm) where the MPC basket flow holes are located is not modeled. The length of the basket not modeled (0.9 inches (2.286 cm)) was determined by calculating the equivalent area removed by the flow holes. This method of approximation is conservative because no material for the basket shielding is provided in the 0.9 inch (2.286 cm) area at the top and bottom of the MPC basket.

2. The upper and lower fuel spacers are not modeled. The fuel spacers are not needed on all fuel assembly types. However, most PWR fuel assemblies will have upper and lower fuel spacers. The positioning of the fuel assembly for the shielding analysis is determined by the fuel spacer length for the design basis fuel assembly type, but the fuel spacer materials are not modeled. This is conservative since it removes steel which would provide a small amount of additional shielding.

3. For the MPC-24, MPC-32 and the MPC-68, the MPC basket supports are not modeled. This is conservative since it removes steel which would provide a small increase in shielding. The aluminum heat conduction elements are also conservatively not modeled.

4. The MPC-24 basket is fabricated from 5/16 inch (0.7938 cm) thick cell plates. It is conservatively assumed for modeling purposes that the structural portion of the MPC-24 basket is uniformly fabricated from 9/32 inch (0.7144 cm) thick steel. The Boral and sheathing are modeled explicitly. This is conservative since it removes steel which would provide a small amount of additional shielding.

5. In the modeling of the BWR fuel assemblies, the zircaloy flow channel was not represented. This was done because it cannot be guaranteed that all BWR fuel assemblies will have an associated flow channel when placed in the MPC. The flow channel does not contribute to the source, but does provide some small amount of shielding. However, no credit is taken for this additional shielding.

6. In the MPC-24, conservatively, all Boral panels on the periphery were modeled with a reduced width of 5 inches (12.7 cm) compared to 6.25 inches (15.875 cm) or 7.5 inches (19.05 cm).

7. The MPC-68 is designed for two lid thicknesses: 9.5 inches (24.13 cm) and 10 inches (25.4 cm). Conservatively, all calculations reported in this chapter were performed with the 9.5 inch (24.13 cm) thick lid.

During this project several design changes occurred that affected the drawings, but did not significantly affect the MCNP models. Therefore, the models may not exactly represent the drawings. The discrepancies between models and drawings are listed and discussed here.

MPC Modeling Discrepancies

1. In the MPCs, there is a sump in the baseplate to enhance draining of the MPC. This localized reduction in the thickness of the baseplate was not modeled. Since there is significant shielding and distance in the HI-STAR outside the MPC baseplate, this localized reduction in shielding will not affect the calculated dose rates outside the HI-STAR.

2. The design configuration of the MPC-24 has been enhanced for criticality purposes. The general location of the 24 assemblies remains basically the same, therefore the shielding analysis continues to use the superseded configuration. Figure 5.3.11 shows the superseded and current configuration for the MPC-24 for comparison.

3. The sheathing thickness on the new MPC-24 configuration was reduced from 0.06 inches (0.1524 cm) to 0.0235 inches (0.0597 cm). However, the model still uses 0.06 inches (0.1524 cm). This discrepancy is compensated for by the use of 9/32 inch (0.7144 cm) cell walls and 5 inch (12.7) boral on the periphery as described above. MCNP calculations were performed with the new MPC-24 configuration under Docket No. 72-1014 in the 100-ton HI-TRAC for comparison to the superceded configuration. These results indicate that on the side of the overpack, the dose rates decrease by approximately 12% on the surface. These results demonstrate that using the superceded MPC-24 design is conservative.

5.3.1.1 Fuel Configuration

As described above, the active fuel region is modeled as a homogenous zone. The end fittings and the plenum regions are also modeled as homogenous regions of steel. The masses of steel used in these regions are shown in Table 5.2.1. The axial description of the design basis fuel assemblies is provided in Table 5.3.1. Figures 5.3.7 and 5.3.8 graphically depict the location of the PWR and BWR fuel assemblies within the HI-STAR 100 System. The axial locations of the Boral, basket, pocket trunnion, and transition areas are shown in these figures.

5.3.1.2 Streaming Considerations

The streaming from the radial steel fins and pocket trunnions in the neutron shield is evaluated in Section 5.4.1. The MCNP model of the HI-STAR 100 completely describes the radial steel fins and pocket trunnions, thereby properly accounting for the streaming effect. This is discussed further in Section 5.4.1.

The design of the HI-STAR 100 System, as described in the drawings in Section 1.5, has eliminated all other possible streaming paths. Therefore, the MCNP model does not represent any additional streaming paths. A brief justification of this assumption is provided for each penetration.

- The lifting trunnions will remain installed in the overpack top flange. No credit is taken for any part of the trunnion that extends outside of the overpack.

- The pocket trunnions are modeled as solid blocks of steel. The pocket trunnion will be filled with a solid steel rotation trunnion attached to the transport frame during handling and a shield plug when located at the ISFSI pad.

- The threaded holes in the MPC lid are plugged with solid plugs during storage and, therefore, do not create a void in the MPC lid.

- The drain and vent ports in the MPC lid are designed to eliminate streaming paths. The steel lost in the MPC lid at the port location is replaced with a block of steel approximately 6 inches (15.24 cm) thick below the port opening and attached to the underside of the lid. This design feature is shown on the drawings in Section 1.5. The MCNP model did not explicitly represent this arrangement but, rather, modeled the MPC lid as a solid piece.

- The penetrations in the overpack are filled with bolts that extend into the penetration when in storage operations, thereby eliminating any potential direct streaming paths. Cover plates are also designed in such a way as to maintain the thickness of the overpack to the maximum extent practical. Therefore, the MCNP model does not represent any streaming paths due to penetrations in the overpack.

5.3.2 Regional Densities

Composition and densities of the various materials used in the HI-STAR 100 System shielding analyses are given in Tables 5.3.2 and 5.3.3. All of the materials and their actual geometries are represented in the MCNP model.

Sections 4.4 and 4.5 demonstrate that all materials used in the HI-STAR 100 System remain below their design temperatures as specified in Table 2.2.3 during all normal conditions.

Therefore, the shielding analysis does not address changes in the material density or composition as a result of temperature changes.

Chapter 11 discusses the effect of the various accident conditions on the temperatures of the shielding materials and the resultant impact on their shielding effectiveness. As stated in Section 5.1.2, there is only one accident that has any significant impact on the shielding configuration. This accident is the loss of the neutron shield in the HI-STAR 100 System as a result of fire or other damage. The change in the neutron shield was conservatively analyzed by assuming that the entire volume of the neutron shield was replaced by void.

As discussed in Chapter 5 of HI-STORM 100 FSAR [5.3.1], approved by the NRC, *the MPCs can be manufactured with one of two possible neutron absorbing materials: Boral or Metamic. Both materials are made of aluminum and* B_4C *powder. The Boral contains an aluminum and* B_4C *powder mixture sandwiched between two aluminum plates while the Metamic is a single plate. The minimum* ^{10}B *areal density is the same for Boral and Metamic while the thicknesses are essentially the same. Therefore, the mass of Aluminum and* B_4C *are essentially equivalent and there is no distinction between the two materials from a shielding perspective. As a result,* Table 5.3.2 *identifies the composition for Boral and no explicit calculations were performed with Metamic*.

Table 5.3.1

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE FUEL ASSEMBLIES†
(The values in parentheses are in cm)

Region	**Start (in.) (cm)**	**Finish (in.) (cm)**	**Length (in.) (cm)**	**Actual Material**	**Modeled Material**
PWR					
Lower End Fitting	0.0	7.375 (18.7325)	7.375 (18.7325)	SS304	SS304
Space	7.375 (18.7325)	8.375 (21.2725)	1.0 (2.54)	zircaloy	void
Fuel	8.375 (21.2725)	152.375 (387.0325)	144 (365.76)	fuel & zircaloy	fuel
Gas Plenum Springs	152.375 (387.0325)	156.1875 (396.7163)	3.8125 (9.6838)	SS304 & zircaloy	SS304
Gas Plenum Spacer	156.1875 (396.7163)	160.5625 (407.8288)	4.375 (11.1125)	SS304 & zircaloy	SS304
Upper End Fitting	160.5625 (407.82875)	165.625 (420.6875)	5.0625 (12.8588)	SS304	SS304
BWR					
Lower End Fitting	0.0	7.385 (18.7579)	7.385 (18.7579)	SS304	SS304
Fuel	7.385 (18.7579)	151.385 (384.5179)	144 (365.76)	fuel & zircaloy	fuel
Space	151.385 (384.5179)	157.385 (399.7579)	6 (15.24)	zircaloy	void
Gas Plenum Springs	157.385 (399.7579)	166.865 (423.8371)	9.48 (24.0792)	SS304 & zircaloy	SS304
Expansion Springs	166.865 (423.8371)	168.215 (427.2661)	1.35 (3.429)	SS304	SS304
Upper End Fitting	168.215 (427.2661)	171.555 (435.7497)	3.34 (8.4836)	SS304	SS304
Handle	171.555 (435.7497)	176 (447.04)	4.445 (11.2903)	SS304	SS304

† All dimensions start at the bottom of the fuel assembly. The length of the lower fuel spacer must be added to the distances to determine the distance from the top of the MPC baseplate.

Table 5.3.2

COMPOSITION OF THE MATERIALS IN THE HI-STAR 100 SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Uranium Oxide	10.412	^{235}U	2.9971(BWR) 3.2615(PWR)
		^{238}U	85.1529(BWR) 84.8885(PWR)
		O	11.85
Boral	2.644	^{10}B	4.4226 (MPC-68) 4.367 (MPC-24 and MPC-32)
		^{11}B	20.1474 (MPC-68) 19.893 (MPC-24 and MPC-32)
		Al	68.61 (MPC-68) 69.01 (MPC-24 and MPC-32)
		C	6.82 (MPC-68) 6.73 (MPC-24 and MPC-32)
SS304	7.92	Cr	19
		Mn	2
		Fe	69.5
		Ni	9.5
Carbon Steel	7.82	C	0.5
		Fe	99.5
Zircaloy	6.55	Zr	100

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STAR 100 SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Neutron Shield Holtite-A	1.61	C	27.66039
		H	5.92
		Al	21.285
		N	1.98
		O	42.372
		^{10}B	0.14087
		^{11}B	0.64174
BWR Fuel Region Mixture	3.979996	^{235}U	2.4483
		^{238}U	69.5601
		O	9.6801
		Zr	18.3115
PWR Fuel Region Mixture	3.853705	^{235}U	2.6944
		^{238}U	70.1276
		O	9.7895
		Zr	17.3885
Lower End Fitting (PWR)	1.0783	SS304	100
Gas Plenum Springs (PWR)	0.1591	SS304	100
Gas Plenum Spacer (PWR)	0.1591	SS304	100

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STAR 100 SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Upper End Fitting (PWR)	1.5410	SS304	100
Lower End Fitting (BWR)	1.5130	SS304	100
Gas Plenum Springs (BWR)	0.2701	SS304	100
Expansion Springs (BWR)	0.6897	SS304	100
Upper End Fitting (BWR)	1.3939	SS304	100
Handle (BWR)	0.2619	SS304	100

Table 5.3.3

COMPOSITION OF THE FUEL IN THE MIXED OXIDE FUEL ASSEMBLIES IN THE MPC-68 OF THE HI-STAR 100 SYSTEM

Component	**Density (g/cm^3)**	**Elements**	**Mass Fraction (%)**
Mixed Oxide Pellets	10.412	^{238}U	84.498
		^{235}U	0.612
		^{238}Pu	0.421
		^{239}Pu	1.455
		^{240}Pu	0.034
		^{241}Pu	0.123
		^{242}Pu	0.007
		O	11.85
Uranium Oxide Pellets	10.412	^{238}U	86.175
		^{235}U	1.975
		O	11.85

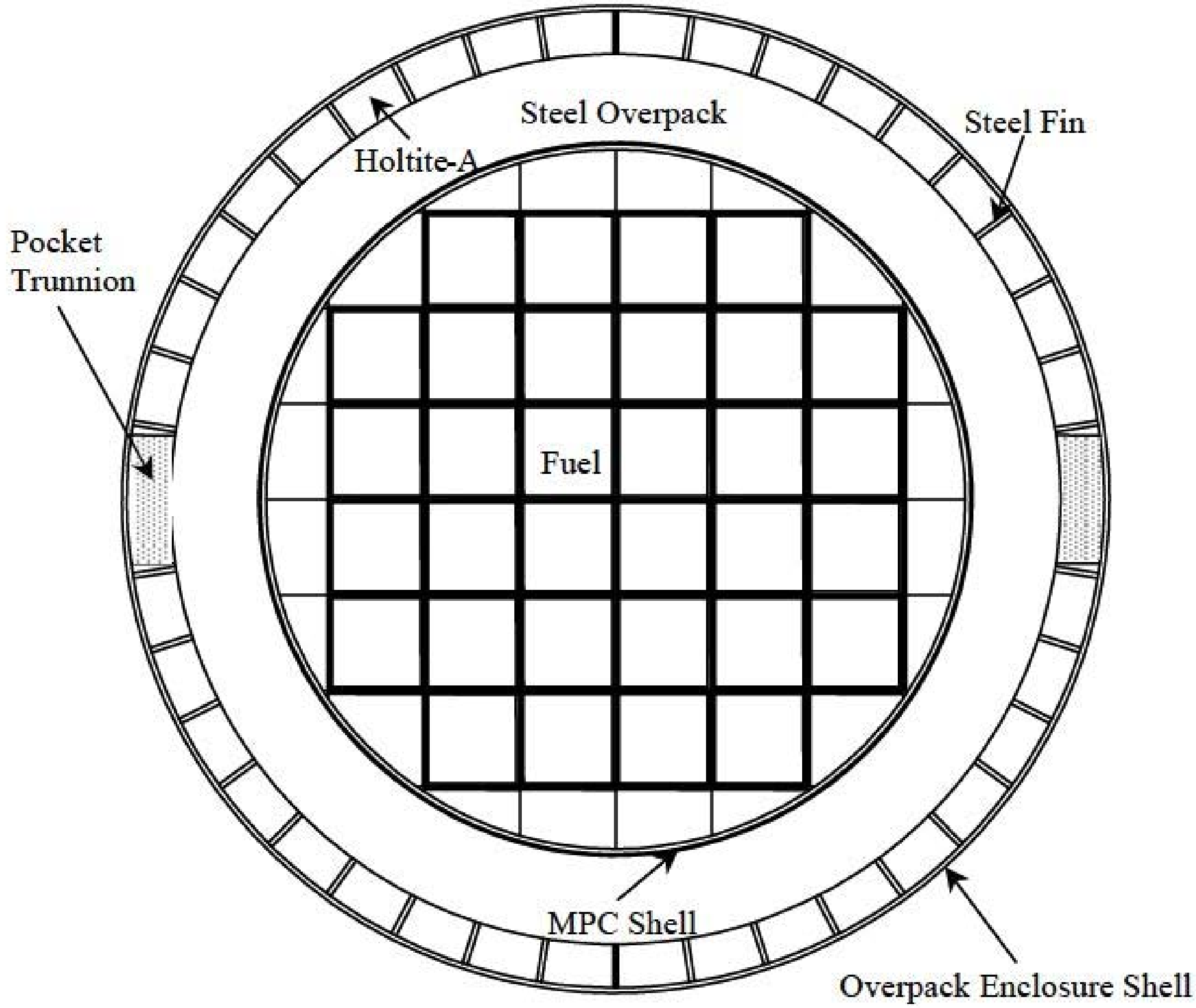


FIGURE 5.3.1; HI-STAR 100 OVERPACK WITH MPC-32 CROSS SECTIONAL VIEW

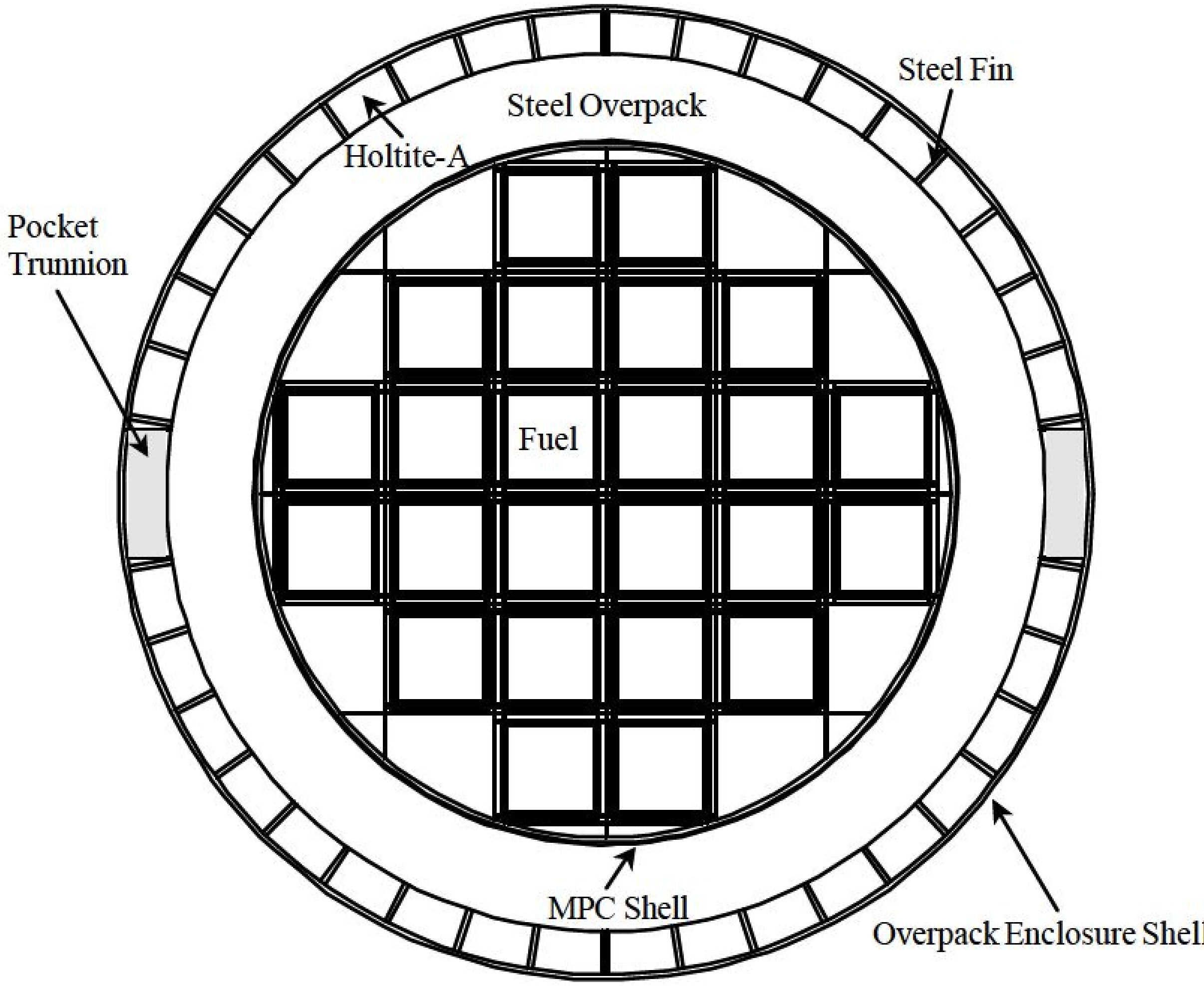


FIGURE 5.3.2; HI-STAR 100 OVERPACK WITH MPC-24 CROSS SECTIONAL VIEW AS MODELLED IN MCNP‡

‡ This figure is drawn to scale using the MCNP plotter.

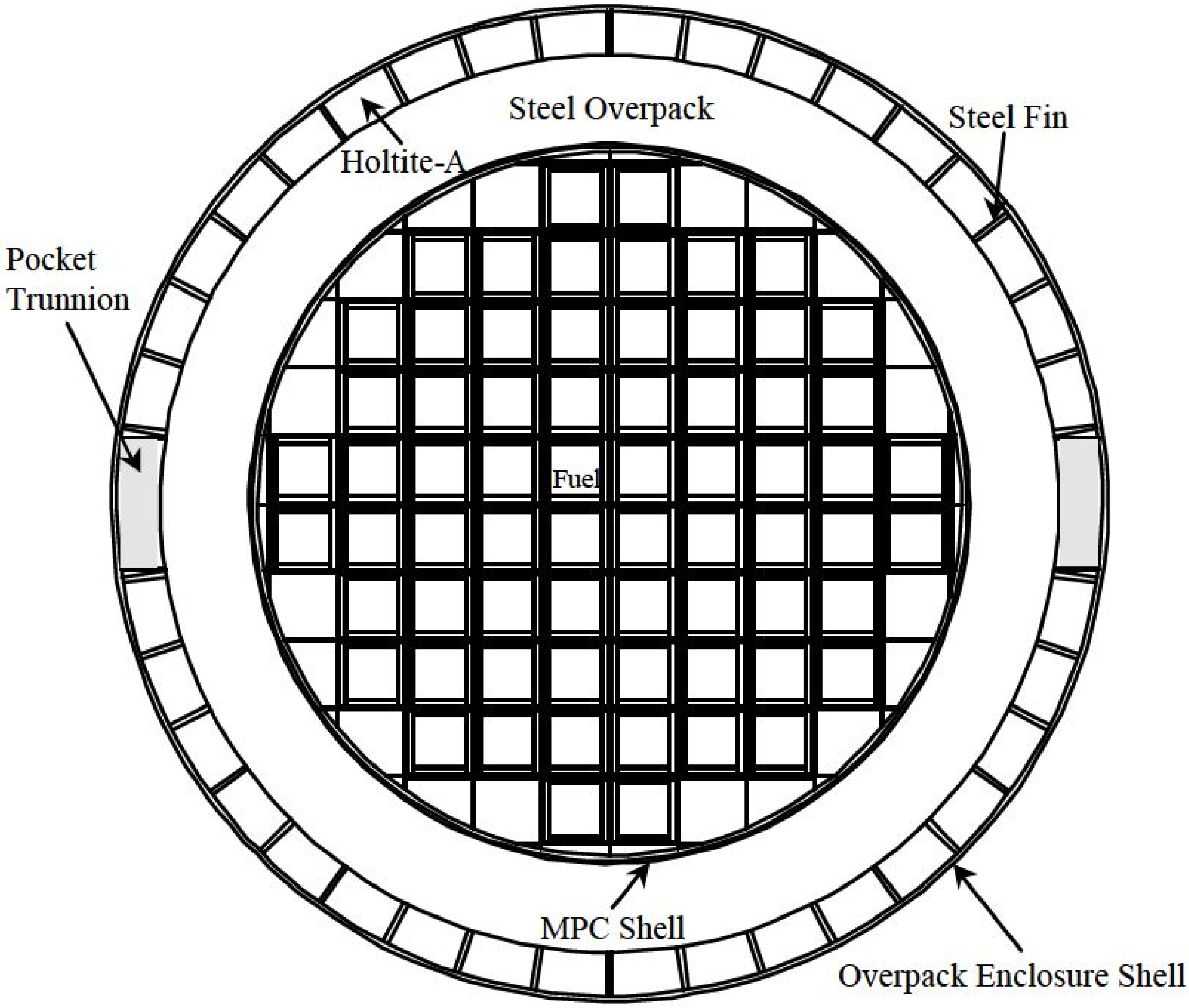


FIGURE 5.3.3; HI-STAR 100 OVERPACK WITH MPC-68 CROSS SECTIONAL VIEW AS MODELLED IN MCNP§

§ This figure is drawn to scale using the MCNP plotter.

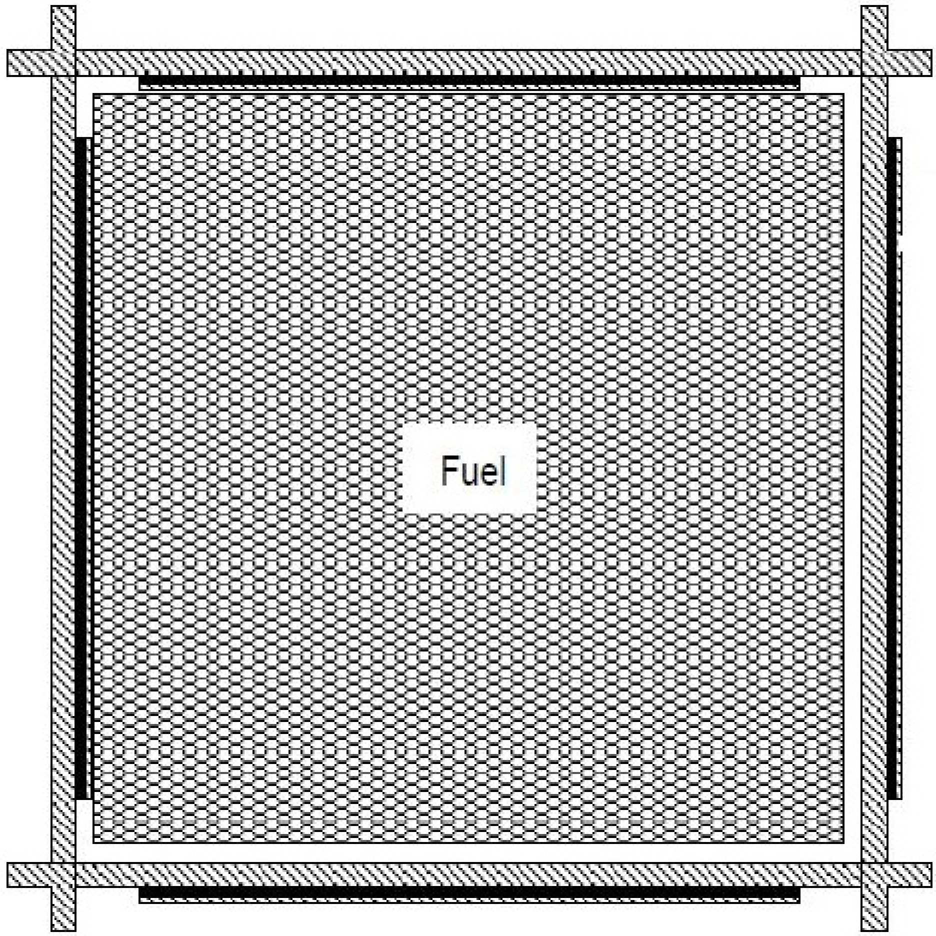


FIGURE 5.3.4; CROSS SECTIONAL VIEW OF AN MPC-32 BASKET CELL

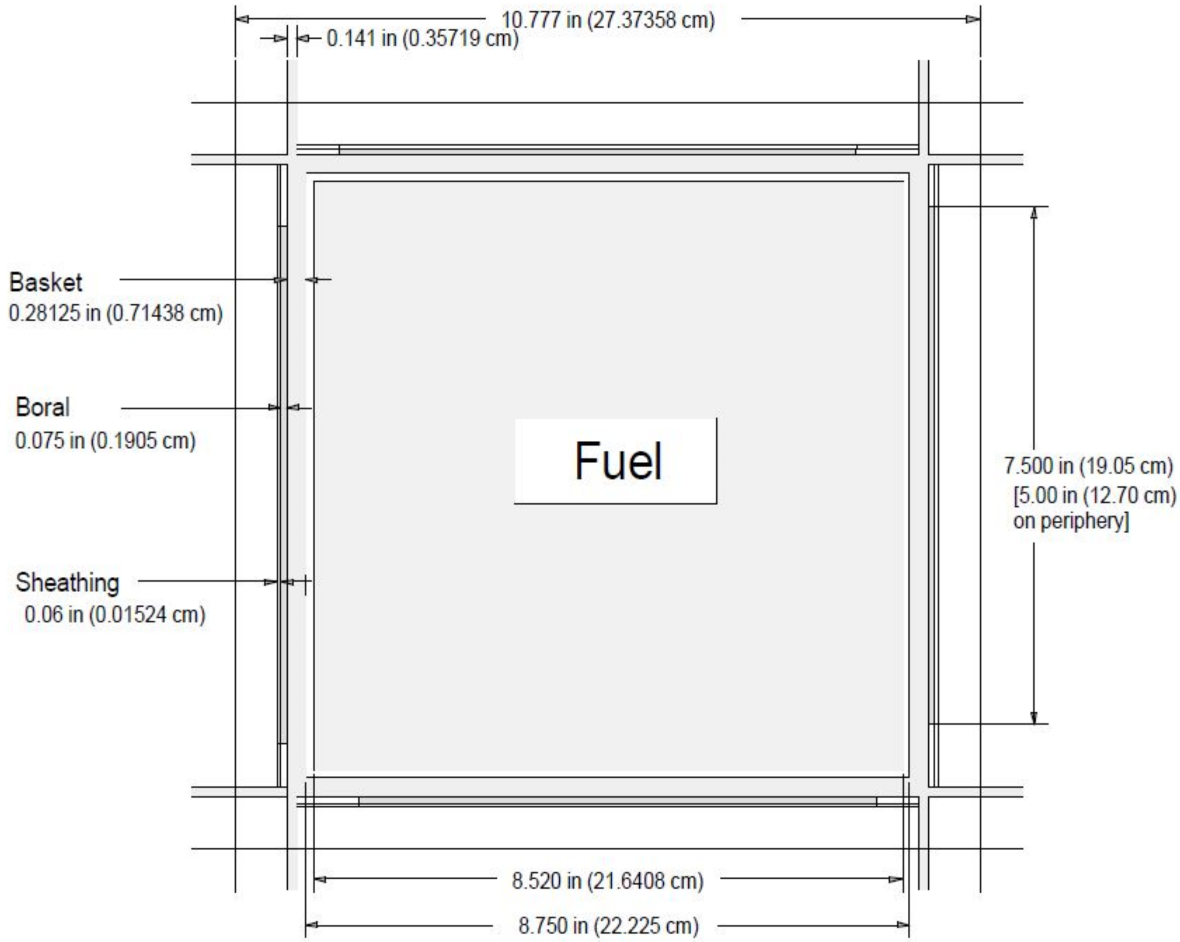


FIGURE 5.3.5; CROSS SECTIONAL VIEW OF AN MPC-24 BASKET CELL AS MODELED IN MCNP

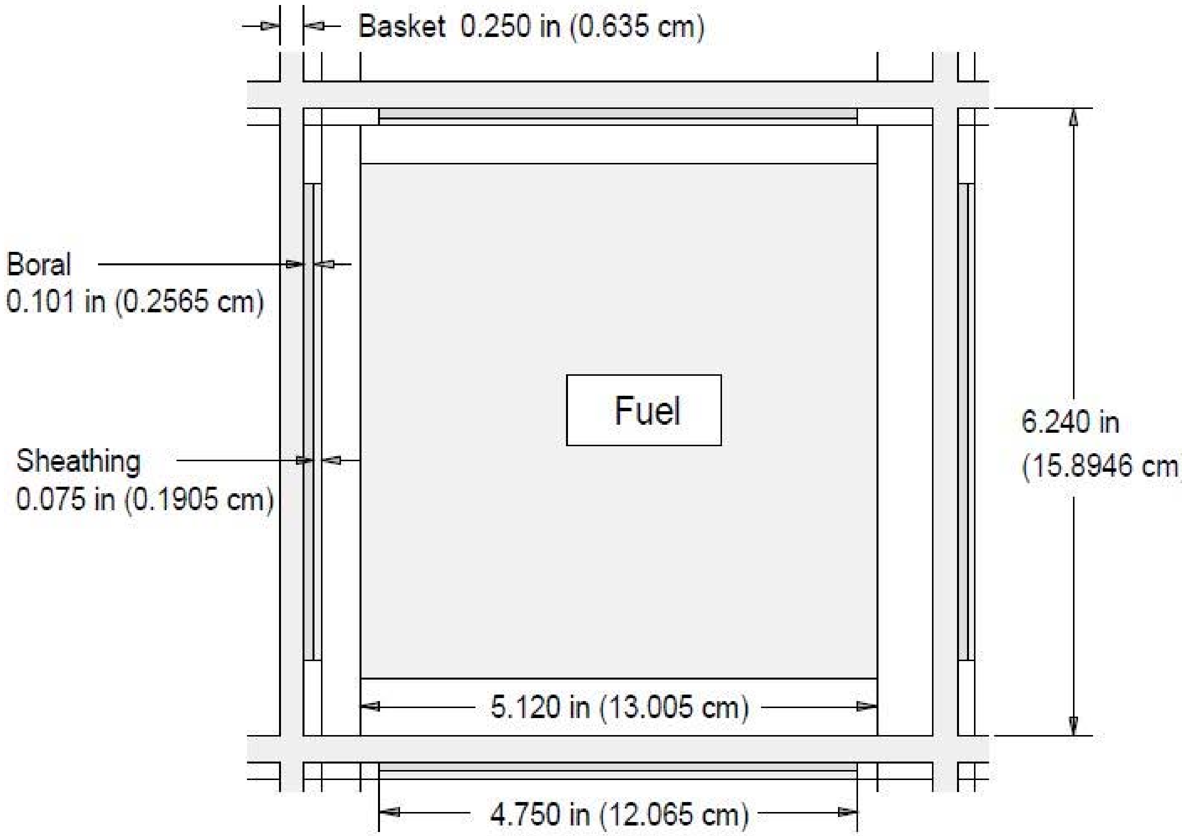


FIGURE 5.3.6; CROSS SECTIONAL VIEW OF AN MPC-68 BASKET CELL AS MODELED IN MCNP

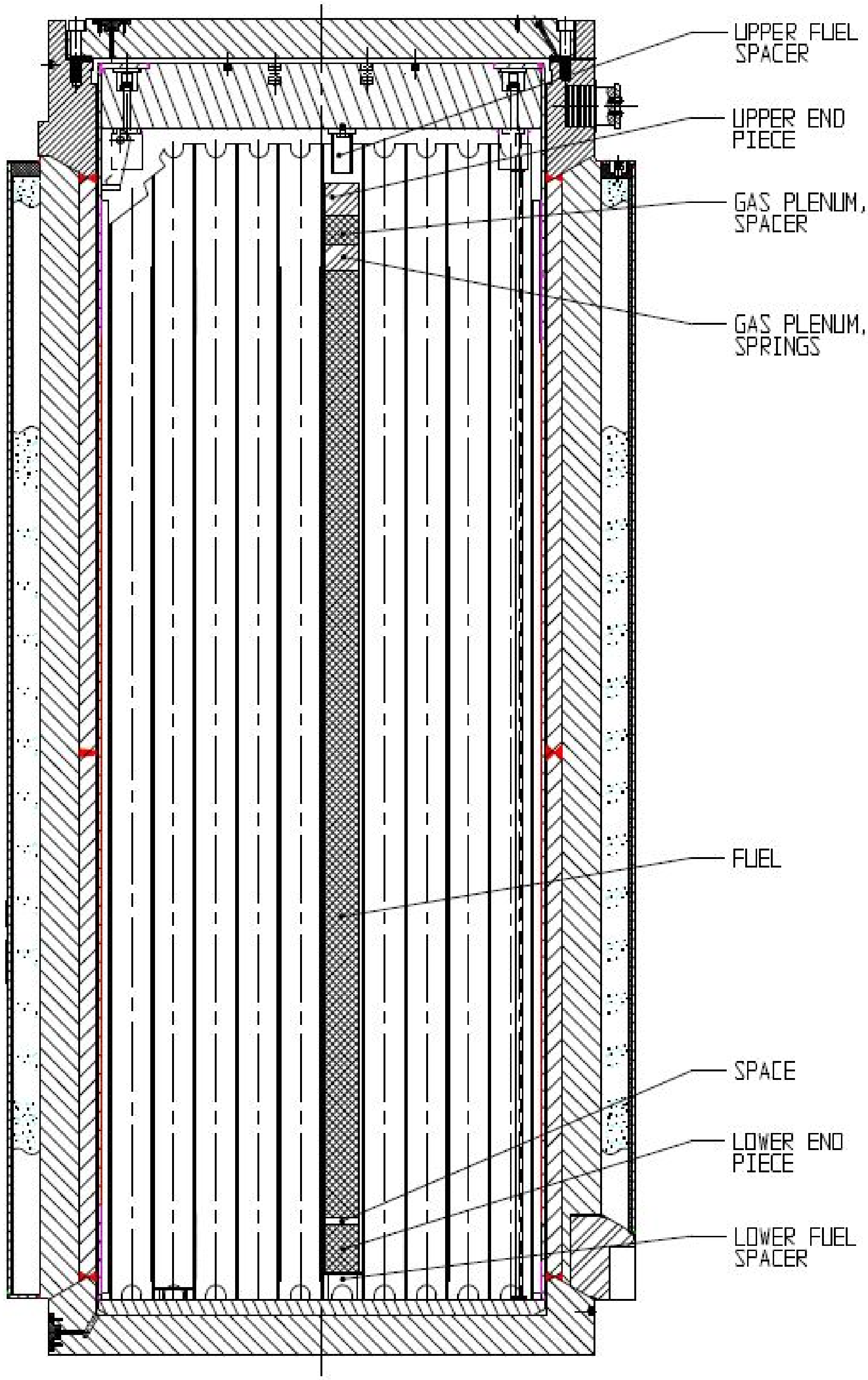


FIGURE 5.3.7; AXIAL LOCATION OF PWR DESIGN BASIS FUEL IN THE HI-STAR 100 SYSTEM

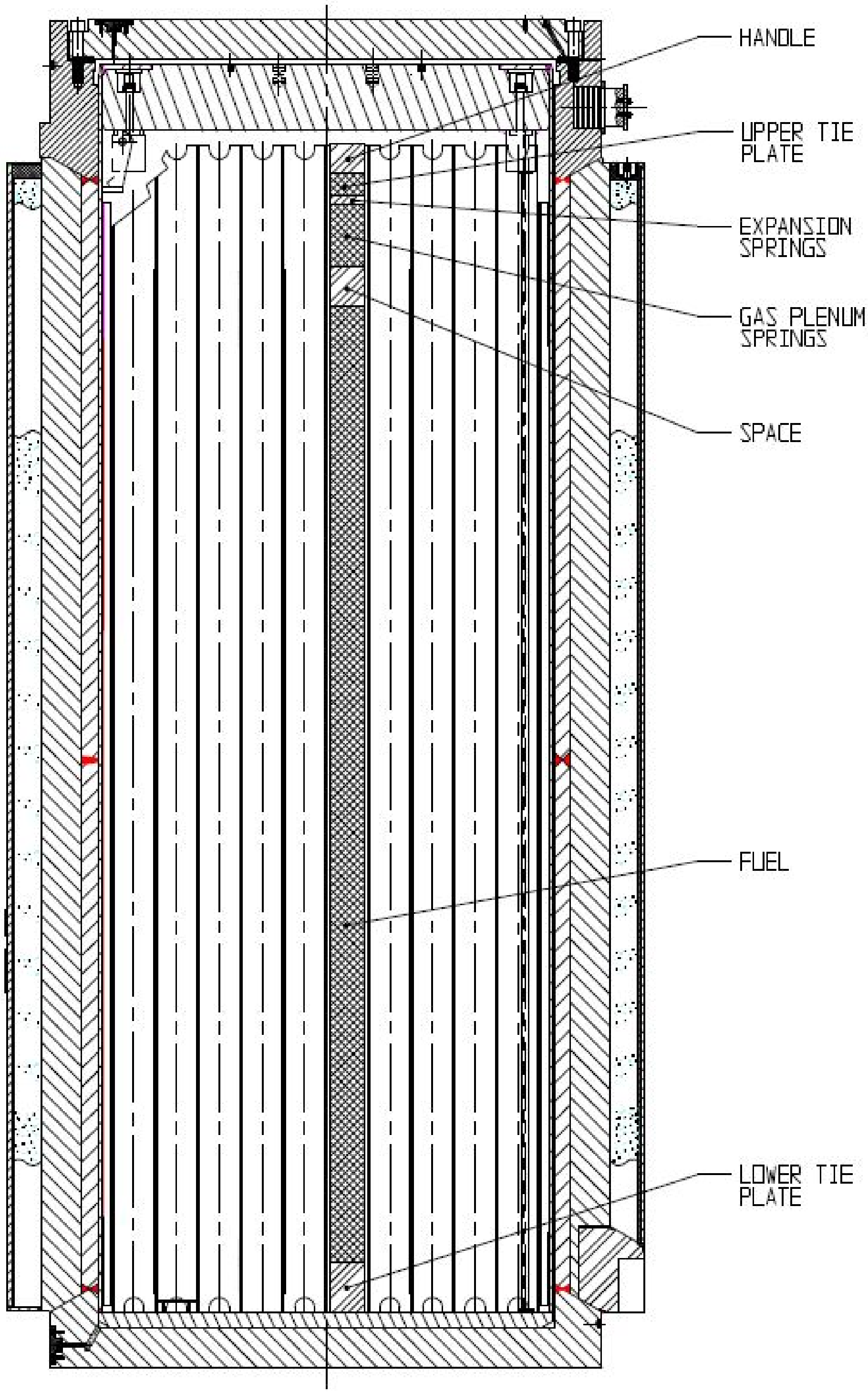


FIGURE 5.3.8; AXIAL LOCATION OF BWR DESIGN BASIS FUEL IN THE HI-STAR 100 SYSTEM

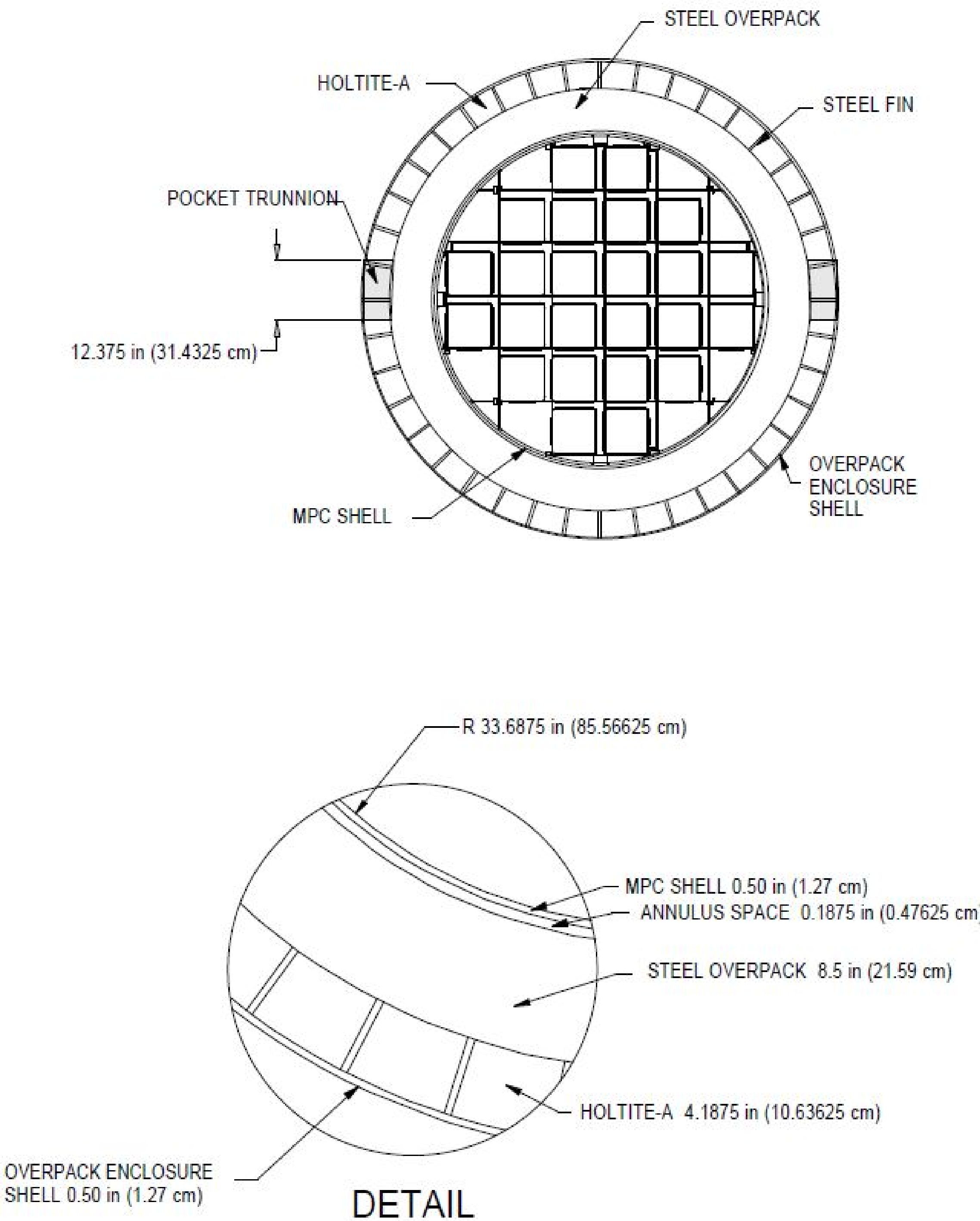


FIGURE 5.3.9; HI-STAR 100 OVERPACK WITH MPC-24 CROSS SECTIONAL VIEW SHOWING THE THICKNESS OF THE MPC SHELL AND OVERPACK AS MODELED IN MCNP

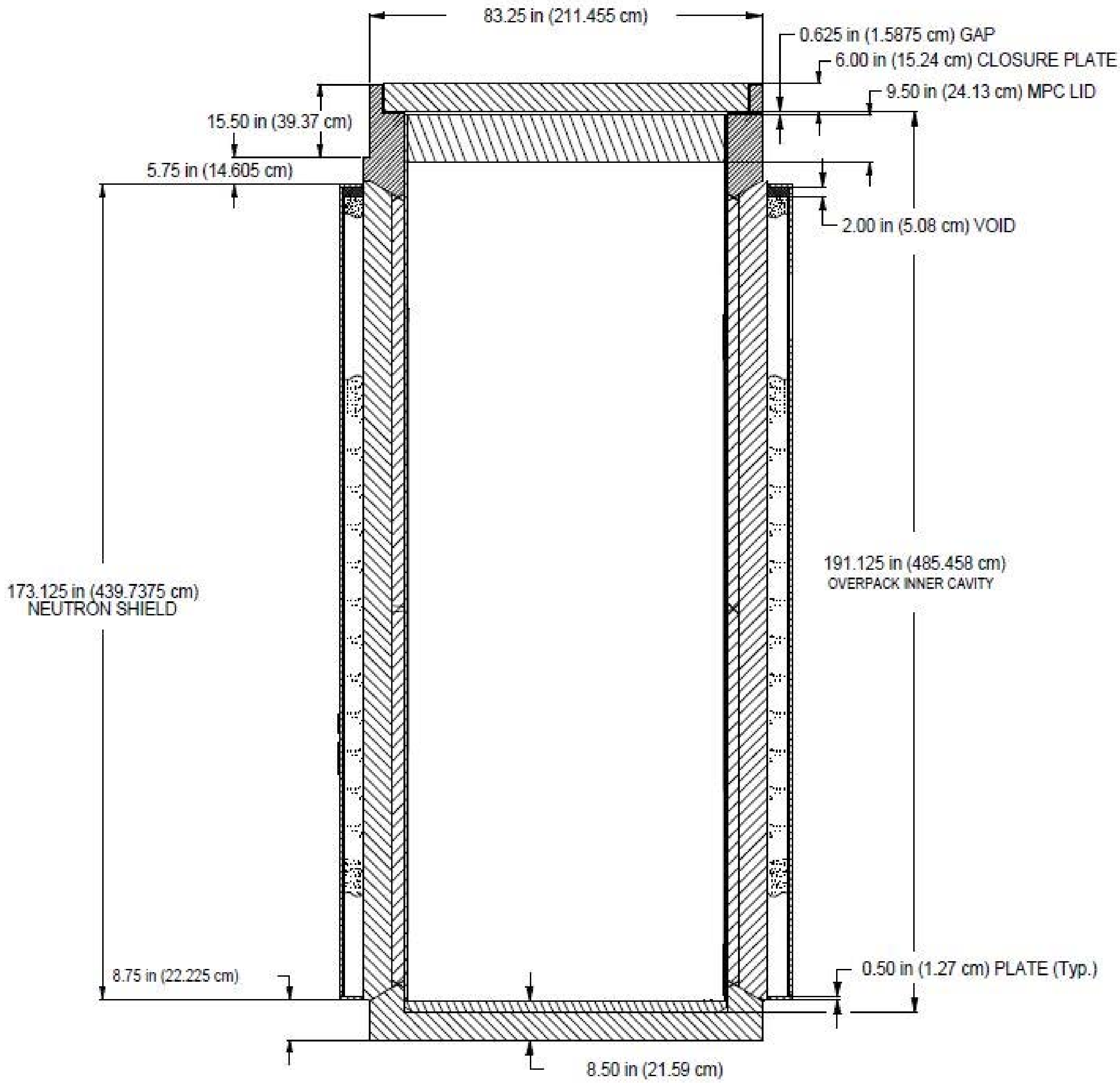
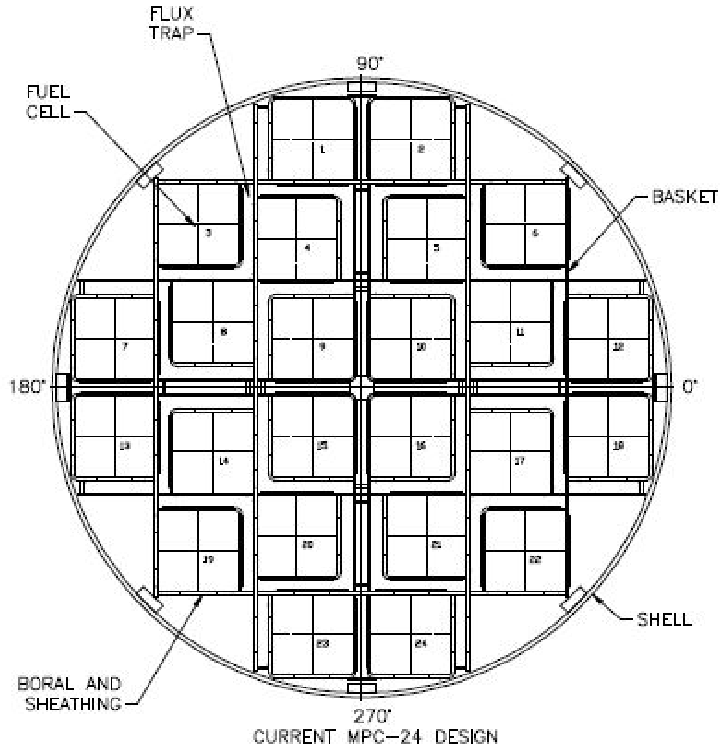


FIGURE 5.3.10; AXIAL VIEW OF HI-STAR 100 OVERPACK AND MPC WITH AXIAL DIMENSIONS SHOWN AS MODELED IN MCNP



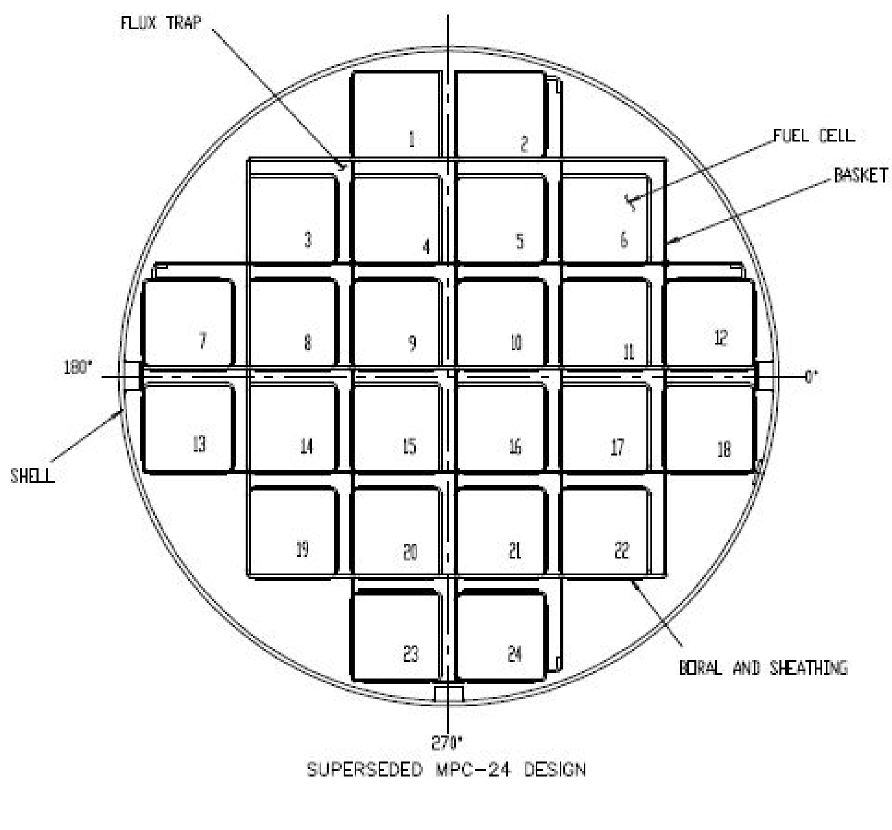


FIGURE 5.3.11; CROSS SECTIONAL VIEWS OF THE CURRENT MPC-24 DESIGN AND THE SUPERSEDED MPC-24 WHICH IS USED IN THE MCNP MODELS.

5.4 SHIELDING EVALUATION

The MCNP-4A code[5.1.1] was used for all of the shielding analyses. MCNP is a continuous energy, three-dimensional, coupled neutron-photon-electron Monte Carlo transport code. Continuous energy cross-section data is represented with sufficient energy points to permit linear-linear interpolation between these points. The individual cross-section libraries used for each nuclide are those recommended by the MCNP manual. All of these data are based on ENDF/B-V data. MCNP has been extensively benchmarked against experimental data by the large user community. References [5.4.2], [5.4.3], and [5.4.4] are three examples of the benchmarking that has been performed.

The energy distribution of the source term, as described earlier, is used explicitly in the MCNP model. A different MCNP calculation is performed for each of the three source terms (neutron, decay gamma, and ^{60}Co). The axial distribution of the fuel source term is described in Table 2.1.8. The PWR and BWR axial burnup distributions were obtained from References [5.4.5] and [5.4.6] respectively. These axial distributions were obtained from operating plants and are representative of PWR and BWR fuel with burnups greater than 30,000 MWD/MTU. The ^{60}Co source in the activated hardware was assumed to be uniformly distributed over the appropriate regions.

It has been shown that the neutron source strength varies as the burnup level raised by the power of 4.2. Since this relationship is non-linear and since the burnup in the axial center of a fuel assembly is greater than the average burnup, the neutron source strength in the axial center of the assembly is greater than the relative burnup times the average neutron source strength. In order to account for this effect, the neutron source strength in each of the 10 axial nodes listed in Table 2.1.8 was determined by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The peak relative burnups listed in Table 2.1.8 for the PWR and BWR fuels are 1.105 and 1.195 respectively. Using the power of 4.2 relationship results in a 37.6% ($1.105^{4.2}/1.105$) and 76.8% ($1.195^{4.2}/1.195$) increase in the neutron source strength in the peak nodes for the PWR and BWR fuel respectively. The total neutron source strength increases by 15.6% for the PWR fuel assemblies and 36.9% for the BWR fuel assemblies.

MCNP was used to calculate dose at the various desired locations. MCNP calculates neutron or photon flux which can be converted into dose by the use of dose response functions. This is done internally in MCNP and the dose response functions are listed in the input file. The response functions used in these calculations are listed in Table 5.4.1 and were taken from ANSI/ANS 6.1.1, 1977 [5.4.1].

Tables 5.4.2 through 5.4.7, and Tables 5.4.18 through 5.4.20 list the normal condition dose rates (from each of the three radiation sources) adjacent to the overpack for each of the burnup levels and cooling times evaluated for the MPC-24,MPC-68 and MPC-32. Tables 5.4.8, 5.4.9 and 5.4.21 provide the total dose rate for each burnup level and cooling time for the MPC-24,MPC-68, and MPC-32, respectively. This information was used to determine the worst case burnup level and cooling time and corresponding maximum dose rates reported in Section 5.1. A detailed discussion of the normal, off-normal, and accident condition dose rates was provided in Sections 5.1.1 and 5.1.2.

A total dose rate comparison between the design basis source term and representative burnup-cooling time combinations for MPC-32 is provided in Table 5.4.24. These representative burnup-cooling time combinations are more realistic, but still they bound the MPC-32 loading patterns provided in Table 2.1.15.

Since MCNP is a statistical code, there is an uncertainty associated with the calculated values. In MCNP the uncertainty is expressed as the relative error which is defined as the standard deviation of the mean divided by the mean. Therefore, the standard deviation is represented as a percentage of the mean. The relative error for the total dose rates presented in this chapter were typically less than 3% and the relative error for the individual dose components was typically less than 5%.

5.4.1 Streaming Through Radial Steel Fins and Pocket Trunnions

The HI-STAR 100 overpack utilizes 0.5 inch (1.27 cm) thick radial steel fins for structural support and cooling. The attenuation of neutrons through steel is substantially less than the attenuation of neutrons through the Holtite-A. Therefore, it is possible to have neutron streaming through the fins which could result in a localized dose peak. The reverse is true for photons which would result in a localized reduction in the photon dose. Analyses were performed to determine the magnitude of the dose peaks and depressions and the impact on localized dose as compared to average total dose. This effect was evaluated at the radial surface of the cask and a distance of one meter from the cask.

In addition to the fins, the pocket trunnions are essentially blocks of steel that are approximately 12 inches (30.48 cm) wide and 12 inches (30.48 cm) high. The effect of the pocket trunnion on neutron streaming and photon transmission will be more substantial than the effect of a single fin. Therefore, analyses were performed to quantity this effect. Figure 5.1.1 illustrates the location of the pocket trunnion and its axial position relative to the active fuel. This position will be important in the discussion that follows.

The effect of streaming through the pocket trunnion and the fins was analyzed using MCNP. The model used was an infinite height radial model which consisted of the MPC and the surrounding overpack. The active fuel region of the fuel assemblies was represented in the MPC basket when the neutron source was used and the lower steel regions of the fuel elements were presented in the MPC basket when the cobalt source was used. The pocket trunnion was represented in this infinite model as being axially adjacent to the active fuel. A calculation was not performed with the photon source. Any depression of the gamma dose due to the steel will be evident when using the cobalt source and this will conservatively bound the effects due to the photon source. This is because the average energy of the photons from ^{60}Co is higher than the average energy of decay gammas.

The MPC-24 and the MPC-68 were analyzed. Figure 5.4.1 shows a quarter of the HI-STAR 100 overpack with 91 azimuthal bins drawn. There is one bin per steel fin and 8 bins in each Holtite-A region. This azimuthal binning structure was used in an infinite height two-dimensional model of the MPC and overpack. The dose was calculated in each of these bins and then compared to the average dose calculated over the surface to determine a peak-to-average ratio for the dose in that bin. The location of the pocket trunnion is shown in Figure 5.4.1. The pocket trunnion was modeled as solid steel. During storage, a shield plug shall be placed in the pocket trunnion recess, and during handling operations a steel rotation trunnion shall be placed in the pocket trunnion recess. To conservatively evaluate the peak-to-average ratio, the pocket trunnion is assumed to be solid steel. The peak-to-average ratio was calculated for the entire pocket trunnion which would correspond to the first seven azimuthal bins.

Table 5.4.10 provides the peak-to-average ratios that were calculated for the various dose components and locations. The peak-to-average ratios were essentially the same for all evaluated MPCs, therefore, only one set of values is shown. The values presented for the pocket trunnions are very conservative since the two-dimensional model represented the trunnion as infinite in height whereas the actual height is approximately 12 inches (30.48 cm). In addition, the pocket trunnion was represented as being axially adjacent to the active fuel which is not completely accurate for the design basis fuel. The infinite two-dimensional model therefore does not represent any leakage out of the pocket trunnion in the axial direction which would reduce the peaking effect.

Table 5.4.11 presents the dose rates at Dose Point #2 (see Figure 5.1.1) and the adjusted dose rates at this point to account for the streaming effects. An additional dose point labeled 2a is listed in this table. This location is axially adjacent to the pocket trunnion and approximately 6 feet (182.88 cm) below Dose Point #2. Based on these results it can be concluded that the streaming effect is noticeable but is not of significant concern.

5.4.2 Damaged Fuel Post-Accident Shielding Evaluation

As discussed in Section 5.2.5.2, the analysis presented below, even though it is for damaged fuel, demonstrates the acceptability of storing intact Humboldt Bay 6x6 and intact Dresden 1 6x6 fuel assemblies.

For the damaged fuel and fuel debris accident condition, it is conservatively assumed that the damaged fuel cladding ruptures and all the fuel pellets fall and collect at the bottom of the damaged fuel container. The inner dimension of the damaged fuel container, specified in the design drawings, and the design basis damaged fuel and fuel debris assembly dimensions in Table 5.2.2 are used to calculate the axial height of the rubble in the damaged fuel container assuming 50% compaction. Neglecting the fuel pellet to cladding inner diameter gap, the volume of cladding and fuel pellets available for deposit is calculated assuming the fuel rods are solid.

Using the volume in conjunction with the damaged fuel container, the axial height of rubble is calculated to be 80 inches (203.2 cm).

Dividing the total fuel gamma source for damaged fuel in Table 5.2.6 by the 80 inch (203.2 cm) rubble height provides a gamma source per inch of 9.68e+10 photon/s. Dividing the total neutron source for damaged fuel in Table 5.2.14 by 80 inches (203.2 cm) provides a neutron source per inch of 2.75e+5 neutron/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of 1.76e+12 photon/s and 5.60e+5 neutron/s. These BWR design basis values were calculated by dividing the total source strengths in Tables 5.2.5 and 5.2.13 by the active fuel length of 144 inches (365.76 cm). Therefore, the design basis damaged fuel assembly is bounded by the design basis intact BWR fuel assembly for accident conditions. No explicit analysis of the damaged fuel dose rates are provided as they are bounded by the intact fuel analysis.

5.4.3 Site Boundary Evaluation

Since NUREG-1536 [5.2.1] states that detailed calculations need not be presented, Chapter 12 assigns ultimate compliance responsibilities to the site licensee. Therefore, this subsection describes, by example, the general methodology for performing site boundary dose calculations. The site-specific fuel characteristics, burnup, cooling time, and the site layout and boundary characteristics would be factored into the evaluation performed by the licensee.

As an example of the methodology, the dose from a single MPC-24 cask and various arrays of MPC-24 casks at a distance greater than 100 meters was evaluated with MCNP. In the model the casks were placed on an infinite slab of concrete to account for earth-shine effects. The atmosphere was represented as dry air at a uniform density corresponding to 20 degrees C. The height of air modeled was 800 meters. This is more than sufficient to properly account for skyshine effects.

The annual dose, assuming 100% occupancy (8760 hours), at 300 meters from one cask is presented in Table 5.4.12 for MPC-24 and in Table 5.4.22 for MPC-32 at the varying maximum burnup and minimum cooling times analyzed. This table indicates that the 40,000 MWD/MTU and 5-year cooling is the bounding case for the evaluated MPC-24 combinations, and the 40,000 MWD/MTU and 8-year cooling is the bounding case for the evaluated MPC-32 combinations.

These tables also indicate that the dose due to neutrons is 21% or more of the total dose. This is an important observation because it implies that simplistic analytical methods such as point kernel techniques may not properly account for the neutron transmission and could lead to low estimates of the site boundary dose.

One of the features of MCNP is the ability to calculate the dose from particles that have passed through certain geometrical regions (referred to as surface or cell flagging). This technique was used to estimate the fraction of the dose at distance from particles, both neutron and gamma, passing through the upper flange region of the overpack. This region is referred to as 3 and 4 on Figure 5.1.1. It was found that, for one cask, approximately 9% of the dose comes from this upper flange region. This is a significant fraction of the total dose and one that is only accounted for using three-dimensional analysis, such as MCNP, which properly includes the effects of neutron and gamma skyshine.

Since the upper flange region is located at the top of the cask, it is reasonable to conclude that this contribution to total dose would be unaffected by placing the cask in an array configuration.

The annual dose, assuming 100% occupancy, at distance from an array of casks was calculated in three steps.

1. The annual dose from the radiation leaving the side of a single HI-STAR 100 overpack was calculated at the distance desired. The side of the HI-STAR 100 overpack is defined as any surface between the bottom of the bottom plate and the top of the closure plate including the upper flange area. Dose value = A.

2. The annual dose from the radiation leaving the top of a single HI-STAR 100 overpack was calculated at the distance desired. The top of the HI-STAR 100 overpack is defined as the top of the closure plate. Dose value = B.

3. The annual dose from the radiation leaving the side of a HI-STAR 100 overpack, when it is in the center of a 3x3 array of casks, was calculated at the distance desired. The casks in the array have a 12 foot pitch (3.6576 m). Dose value = C.

The annual dose calculated in each of these three steps was averaged over a cylindrical surface at various distances from the source cask for ease of calculation. In step 3, the dose at the cylindrical surface included contributions from radiation that traveled between the surrounding casks and from radiation that traveled above the surrounding casks and scattered in air to reach the dose location. Therefore, the average dose values from step 3 include all possible paths for radiation to reach the dose location. The values from step 3 represent the dose from a cask in the second row of an array which is shielded by casks in the front row.

The doses calculated in the steps above are listed in Table 5.4.13 for 40,000 MWD/MTU and 5-year cooling for MPC-24, and in Table 5.4.23 for 40,000 MWD/MTU and 8-year cooling for MPC-32. Using these values, the annual dose (at the center of the long side) from an arbitrary 2 by Z array of HI-STAR 100 overpacks can easily be calculated. The following formula describes the method.

Z = number of casks along long side

Dose = ZA + 2ZB + ZC

As an example, the dose from a 2x3 array at 250 meters is presented.

1. The annual dose from the side of a single cask: Dose A = 24.53
2. The annual dose from the top of a single cask: Dose B = 0.63
3. The annual dose from the side of a cask in the center of a 3x3 array: Dose C = 8.81

Using the formula shown above (Z=3) the total dose at 250 meters from a 2x3 array of filled HI-STAR 100 overpacks with MPC-24 is 103.80 mrem/year (1.0380 mSv/year), assuming 100% occupancy.

An important point to notice here is that the dose from the side of the back row of casks is approximately 25% of the total dose. This is a significant contribution and one that would probably not be accounted for properly by simpler methods of analysis.

The results for various arrays of filled HI-STAR 100 overpacks can be found in Section 5.1.1.

5.4.4 Mixed Oxide Fuel Evaluation

The source terms calculated for the Dresden 1 GE 6x6 MOX fuel assemblies can be compared to the design basis source terms for the GE 7x7 assemblies which demonstrates that the MOX fuel source terms are bounded by the design basis source terms and no additional shielding analysis is needed.

Since the active fuel length of the MOX fuel assemblies is shorter than the active fuel length of the design basis fuel, the source terms must be compared on a per inch basis. Dividing the total fuel gamma source for the MOX fuel in Table 5.2.16 by the 110 inch (279.4 cm) active fuel height provides a gamma source per inch of 6.97e+10 photons/s. Dividing the total neutron source for the MOX fuel assemblies in Table 5.2.17 by 110 inches (279.4 cm) provides a neutron source strength per inch of 3.06e+5 neutrons/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of 1.76e+12 photons/s and 5.60e+5 neutrons/s. These BWR design basis values were calculated by dividing the total source strength in Tables 5.2.5 and 5.2.13 by the active fuel length of 144 inches (365.76 cm). This comparison shows that the MOX fuel source terms are bound by the design basis source terms. Therefore, no explicit analysis of dose rates is provided for MOX fuel.

Since the MOX fuel assemblies are Dresden 1 6x6 assemblies, they can also be considered as damaged fuel or fuel debris. Using the same methodology as described in Section 5.4.2, the source term for the MOX fuel is calculated on a per inch basis assuming a post accident rubble height of 80 inches (203.2 cm). The resulting gamma and neutron source strengths are 9.59e+10 photons/s and 4.21e+5 neutrons/s. These values are also bounded by the design basis fuel gamma source per inch and neutron source per inch. Therefore, no explicit analysis of dose rates is provided for MOX fuel in a post accident configuration.

5.4.5 Stainless Steel Clad Fuel Evaluation

Table 5.4.14 presents the dose rates at the center of the HI-STAR 100 overpack, adjacent and at one meter distance, for the stainless steel clad fuel. These dose rates, when compared to Tables 5.1.2, 5.1.3, 5.1.5, and 5.1.6, are very close to the dose rates from the design basis zircaloy clad fuel indicating that these fuel assemblies are acceptable for storage.

As described in Section 5.2.3, it would be incorrect to compare the total source strength from the stainless steel clad fuel assemblies to the source strength from the design basis zircaloy clad fuel assemblies since these assemblies do not have the same active fuel length and since there is a significant gamma source from Cobalt-60 activation in the stainless steel. Therefore it is necessary to calculate the dose rates from the stainless steel clad fuel and compare them to the dose rates from the zircaloy clad fuel. In calculating the dose rates, the source term for the stainless steel fuel was calculated with an artificial active fuel length of 144 inches (365.76 cm) to permit a simple comparison of dose rates from stainless steel clad fuel and zircaloy clad fuel at the center of the HI-STAR 100 overpack. Since the true active fuel length is shorter than 144 inches (365.76 cm) and since the end fitting masses of the stainless steel clad fuel are assumed to be identical to the end fitting masses of the zircaloy clad fuel, the dose rates at the other locations on the overpack are bounded by the dose rates from the design basis zircaloy clad fuel, and therefore, no additional dose rates are presented.

5.4.6 BPRAs and TPDs

In order to verify that the BPRAs and TPDs do not affect the shielding analysis, the total dose rates were calculated for the HI-STAR 100 with MPC-24 assuming all fuel assemblies in the MPC contained either BPRAs or TPDs. For this calculation, three separate burnups, slightly higher than the allowable burnups listed in Table 2.1.13 were used with the corresponding cooling time. Tables 5.4.16 and 5.4.17 present the comparison of the total dose rates around the HI-STAR 100 overpack for PWR fuel with and without BPRAs or TPDs. The design basis dose rates are provided in these tables for easy comparison. A comparison of accident condition dose rates is only performed for assemblies with BPRAs since the TPDs, which are in the upper portion of the fuel assembly, will not have a noticeable impact on the accident dose rates at the centerline of the overpack. These tables illustrate that the dose rates for fuel assemblies containing BPRAs and TPDs are bounded by the design basis 40,000 MWD/MTU and 5 year cooling dose rates listed in Section 5.1.1 and Section 5.1.2. Therefore, the addition of BPRAs and TPDs is bounded by the shielding analysis presented in this chapter.

5.4.7 Dresden Unit 1 Antimony-Beryllium Neutron Sources

Dresden Unit 1 has antimony-beryllium neutron sources which are placed in the water rod location of their fuel assemblies. These sources are steel rods which contain a cylindrical antimony-beryllium source which is 77.25 inches (196.215 cm) in length. The steel rod is approximately 95 inches (241.3 cm) in length. Information obtained from Dresden Unit 1 characterizes these sources in the following manner: "About one-quarter pound of beryllium will be employed as a special neutron source material. The beryllium produces neutrons upon gamma irradiation. The gamma rays for the source at initial start-up will be provided by neutron-activated antimony (about 865 curies (32 TBq)). The source strength is approximately 1E+8 neutrons/second."

As stated above, beryllium produces neutrons through gamma irradiation and in this particular case antimony is used as the gamma source. The threshold gamma energy for producing neutrons from beryllium is 1.666 MeV. The outgoing neutron energy increases as the incident gamma energy increases. Sb-124, which decays by Beta decay with a half life of 60.2 days, produces a gamma of energy 1.69 MeV which is just energetic enough to produce a neutron from beryllium. Approximately 54% of the Beta decays for Sb-124 produce gammas with energies greater than or equal to 1.69 MeV. Therefore, the neutron production rate in the neutron source can be specified as 5.8E-6 neutrons per gamma (1E+8/865/3.7e+10/0.54) with energy greater than 1.666 MeV or 1.16E+5 neutrons/curie (1E+8/865) of Sb-124.

With the short half life of 60.2 days all of the initial Sb-124 is decayed and any Sb-124 that was produced while the neutron source was in the reactor is also decayed since these neutron sources are assumed to have the same minimum cooling time as the Dresden 1 fuel assemblies (array classes 6x6A, 6x6B, 6x6C, and 8x8A) of 18 years. Therefore, there are only two possible gamma sources which can produce neutrons from this antimony-beryllium source. The first is the gammas from the decay of fission products in the fuel assemblies in the MPC. The second gamma source is from Sb-124 which is being produced in the MPC from neutron activation from neutrons from the decay of fission products.

MCNP calculations were performed to determine the gamma source as a result of decay gammas from fuel assemblies and Sb-124 activation. The calculations explicitly modeled the 6x6 fuel assembly described in Table 5.2.2. A single fuel rod was removed and replaced by a guide tube. In order to determine the amount of Sb-124 that is being activated from neutrons in the MPC it was necessary to estimate the amount of antimony in the neutron source. The O.D. of the source was assumed to be the I.D. of the steel rod encasing the source (0.345 in. (0.876 cm)). The length of the source is 77.25 inches (196.215 cm). The beryllium is assumed to be annular in shape encompassing the antimony. Using the assumed O.D. of the beryllium and the mass and length, the I.D. of the beryllium was calculated to be 0.24 inches (0.61 cm). The antimony is assumed to be a solid cylinder with an O.D. equal to the I.D. of the beryllium. These assumptions are conservative since the antimony and beryllium are probably encased in another material which would reduce the mass of antimony. A larger mass of antimony is conservative since the calculated activity of Sb-124 is directly proportional to the initial mass of antimony.

The number of gammas from fuel assemblies with energies greater than 1.666 MeV entering the 77.25 inch (196.215 cm) long neutron source was calculated to be 1.04E+8 gammas/sec which would produce a neutron source of 603.2 neutrons/sec (1.04E+8 * 5.8E-6). The steady state amount of Sb-124 activated in the antimony was calculated to be 39.9 curies (1.476 TBq). This activity level would produce a neutron source of 4.63E+6 neutrons/sec (39.9 * 1.16E+5) or 6.0E+4 neutrons/sec/inch† (2.36E+4 neutrons/sec/cm). These calculations conservatively neglect the reduction in antimony and beryllium which would have occurred while the neutron sources were in the core and being irradiated at full reactor power.

Since this is a localized source (77.25 inches (196.215 cm) in length) it is appropriate to compare the neutron source per inch from the design basis Dresden Unit 1 fuel assembly, 6x6, containing an Sb-Be neutron source to the design basis fuel neutron source per inch. This comparison, presented in Table 5.4.15, demonstrates that a Dresden Unit 1 fuel assembly containing an Sb-Be neutron source is bounded by the design basis fuel.

As stated above, the Sb-Be source is encased in a steel rod. Therefore, the gamma source from the activation of the steel was considered assuming a burnup of 120,000 MWD/MTU which is the maximum burnup assuming the Sb-Be source was in the reactor for the entire 18 year life of Dresden Unit 1. The cooling time assumed was 18 years which is the minimum cooling time for Dresden Unit 1 fuel. The source from the steel was bounded by the design basis fuel assembly. In conclusion, storage of a Dresden Unit 1 Sb-Be neutron source in a Dresden Unit 1 fuel assembly is acceptable and bounded by the current analysis.

5.4.8 Thoria Rod Canister

Based on a comparison of the gamma spectra from Tables 5.2.32 and 5.2.6 for the thoria rod canister and design basis 6x6 fuel assembly, respectively, it is difficult to determine if the thoria rods will be bounded by the 6x6 fuel assemblies. However, it is obvious that the neutron spectra from the 6x6, Table 5.2.14, bounds the thoria rod neutron spectra, Table 5.2.33, with a significant margin. In order to demonstrate that the gamma spectrum from the single thoria rod canister is bounded by the gamma spectrum from the design basis 6x6 fuel assembly, the gamma dose rate on the outer radial surface of the overpack was estimated conservatively assuming an MPC full of thoria rod canisters. This gamma dose rate was compared to an estimate of the dose rate from an MPC full of design basis 6x6 fuel assemblies. The gamma dose rate from the 6x6 fuel was higher than the dose rate from an MPC full of thoria rod canisters. This in conjunction with the significant margin in neutron spectrum and the fact that there is only one thoria rod canister clearly demonstrates that the thoria rod canister is acceptable for storage in the MPC-68 or the MPC-68F.

5.4.9 PWR Fuel Assembly with Non-zircaloy Grid Spacers

PWR fuel assemblies with non-zircaloy (inconel or steel) grid spacers are qualified to be loaded to MPC-32 basket. Since the mass of the spacers is significant and since the cobalt impurity level assumed for inconel is relatively high, the Cobalt-60 activity from the incore spacers may contribute significantly to the external dose rate. As a result, separate burnup and cooling times

† 6.0E+4 = 4.63E+6 / 77.25

were developed for PWR assemblies that utilize zircaloy and non-zircaloy incore spacers. Since steel has a lower cobalt impurity level than inconel, any zircaloy clad PWR assemblies with stainless steel grid spacers are bounded by the analysis performed in this chapter utilizing inconel grid spacers.

The description of the analyzed PWR non-zircaloy grid spacers is provided in Table 5.4.25, including the grid spacer mass and cobalt impurity level in the inconel. These values are very conservative.

The analyzed fuel assembly burnup and compositions, along with surface and 1-m dose rates, are provided in Table 5.4.24. These representative burnup-cooling time combinations bound the MPC-32 loading patterns provided in Table 2.1.15 for non-zircaloy grid spacers (i.e., for the same cooling time, the burnup of the analyzed combinations bound those is Table 2.1.15).

Table 5.4.1

FLUX-TO-DOSE CONVERSION FACTORS
(FROM [5.4.1])

Gamma Energy (MeV)	(rem/hr)/ (photon/cm^2-s)	(Sv/hr)/ (photon/cm^2-s)
0.01	3.96E-06	3.96E-08
0.03	5.82E-07	5.82E-09
0.05	2.90E-07	2.90E-09
0.07	2.58E-07	2.58E-09
0.1	2.83E-07	2.83E-09
0.15	3.79E-07	3.79E-09
0.2	5.01E-07	5.01E-09
0.25	6.31E-07	6.31E-09
0.3	7.59E-07	7.59E-09
0.35	8.78E-07	8.78E-09
0.4	9.85E-07	9.85E-09
0.45	1.08E-06	1.08E-08
0.5	1.17E-06	1.17E-08
0.55	1.27E-06	1.27E-08
0.6	1.36E-06	1.36E-08
0.65	1.44E-06	1.44E-08
0.7	1.52E-06	1.52E-08
0.8	1.68E-06	1.68E-08
1.0	1.98E-06	1.98E-08
1.4	2.51E-06	2.51E-08
1.8	2.99E-06	2.99E-08
2.2	3.42E-06	3.42E-08

Table 5.4.1 (continued)

FLUX-TO-DOSE CONVERSION FACTORS
(FROM [5.4.1])

Gamma Energy (MeV)	(rem/hr)/ (photon/cm^2-s)	(Sv/hr)/ (photon/cm^2-s)
2.6	3.82E-06	3.82E-08
2.8	4.01E-06	4.01E-08
3.25	4.41E-06	4.41E-08
3.75	4.83E-06	4.83E-08
4.25	5.23E-06	5.23E-08
4.75	5.60E-06	5.60E-08
5.0	5.80E-06	5.80E-08
5.25	6.01E-06	6.01E-08
5.75	6.37E-06	6.37E-08
6.25	6.74E-06	6.74E-08
6.75	7.11E-06	7.11E-08
7.5	7.66E-06	7.66E-08
9.0	8.77E-06	8.77E-08
11.0	1.03E-05	1.03E-07
13.0	1.18E-05	1.18E-07
15.0	1.33E-05	1.33E-07

Table 5.4.1 (continued)

FLUX-TO-DOSE CONVERSION FACTORS
(FROM [5.4.1])

Neutron Energy (MeV)	Quality Factor	(rem/hr)†/(n/cm^2-s)	(Sv/hr)† /(n/cm^2-s)
2.5E-8	2.0	3.67E-6	3.67E-08
1.0E-7	2.0	3.67E-6	3.67E-08
1.0E-6	2.0	4.46E-6	4.46E-08
1.0E-5	2.0	4.54E-6	4.54E-08
1.0E-4	2.0	4.18E-6	4.18E-08
1.0E-3	2.0	3.76E-6	3.76E-08
1.0E-2	2.5	3.56E-6	3.56E-08
0.1	7.5	2.17E-5	2.17E-07
0.5	11.0	9.26E-5	9.26E-07
1.0	11.0	1.32E-4	1.32E-06
2.5	9.0	1.25E-4	1.25E-06
5.0	8.0	1.56E-4	1.56E-06
7.0	7.0	1.47E-4	1.47E-06
10.0	6.5	1.47E-4	1.47E-06
14.0	7.5	2.08E-4	2.08E-06
20.0	8.0	2.27E-4	2.27E-06

† Includes the Quality Factor.

Table 5.4.2

DOSE RATES FROM FUEL GAMMAS
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP AND COOLING TIMES†
(The values in parentheses are in mSv/h)

Dose Point†† Location	**40,000 MWD/MTU 5-Year Cooling (mrem/hr) (mSv/h)**	**47,500 MWD/MTU 8-Year Cooling (mrem/hr) (mSv/h)**
1	12.45 (0.1245)	6.93 (0.0693)
2	96.88 (0.9688)	54.98 (0.5498)
3	3.51 (0.0351)	2.20 (0.0220)
4	1.81 (0.0181)	1.11 (0.0111)
5	0.34 (0.0034)	0.42 (0.0042)
6 (dry MPC)†††	27.07 (0.2707)	14.36 (0.1436)
7 (no temp. shield)	100.36 (1.0036)	50.68 (0.5068)
7 (with temp. shield)	28.27 (0.2827)	19.59 (0.1959)

† Gammas generated by neutron capture are included with fuel gammas.

†† Refer to Figure 5.1.1.

††† Overpack closure plate not present.

Table 5.4.3

DOSE RATES FROM ^{60}Co GAMMAS
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP AND COOLING TIMES
(The values in parentheses are in mSv/h)

Dose Point† Location	**40,000 MWD/MTU 5-Year Cooling (mrem/hr) (mSv/h)**	**47,500 MWD/MTU 8-Year Cooling (mrem/hr) (mSv/h)**
1	231.52 (2.3152)	176.39 (1.7639)
2	0.03 (0.0003)	0.02 (0.0002)
3	81.12 (0.8112)	61.80 (0.6180)
4	35.86 (0.3586)	27.32 (0.2732)
5	0.69 (0.0069)	0.53 (0.0053)
6 (dry MPC)††	286.19 (2.8619)	218.05 (2.1805)
7 (no temp. shield)	1432.28 (14.3228)	1091.26 (10.9126)
7 (with temp. shield)	329.84 (3.2984)	251.30 (2.5130)

† Refer to Figure 5.1.1.

†† Overpack closure plate not present.

Table 5.4.4

DOSE RATES FROM NEUTRONS
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP
AND COOLING TIMES
(The values in parentheses are in mSv/h)

Dose Point† Location	**40,000 MWD/MTU 5-Year Cooling (mrem/hr) (mSv/h)**	**47,500 MWD/MTU 8-Year Cooling (mrem/hr) (mSv/h)**
1	82.27 (0.8227)	132.74 (1.3274)
2	22.12 (0.2212)	35.69 (0.3569)
3	70.28 (0.7028)	113.40 (1.1340)
4	39.47 (0.3947)	63.68 (0.6368)
5	56.70 (0.5670)	91.48 (0.9148)
6 (dry MPC)††	126.02 (1.2602)	203.29 (2.0329)
7 (no temp. shield)	397.30 (3.9730)	641.02 (6.4102)
7 (with temp. shield)	19.84 (0.1984)	32.01 (0.3201)

† Refer to Figure 5.1.1.

†† Overpack closure plate not included.

Table 5.4.5

DOSE RATES FROM FUEL GAMMAS
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP AND COOLING TIMES†
(The values in parentheses are in mSv/h)

Dose Point†† Location	35,000 MWD/MTU 5-Year Cooling (mrem/hr) (mSv/h)	45,000 MWD/MTU 9-Year Cooling (mrem/hr) (mSv/h)
1	10.26 (0.1026)	5.47 (0.0547)
2	100.42 (1.0042)	52.26 (0.5226)
3	0.97 (0.0097)	0.81 (0.0081)
4	0.44 (0.0044)	0.37 (0.0037)
5	0.13 (0.0013)	0.20 (0.0020)
6 (dry MPC)†††	9.32 (0.0932)	4.52 (0.0452)
7 (no temp. shield)	64.46 (0.6446)	31.28 (0.3128)
7 (with temp. shield)	20.36 (0.2036)	16.04 (0.1604)

† Gammas generated by neutron capture are included with fuel gammas.

†† Refer to Figure 5.1.1.

††† Overpack closure plate not included.

Table 5.4.6

DOSE RATES FROM ^{60}Co GAMMAS
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP
AND COOLING TIMES
(The values in parentheses are in mSv/h)

Dose Point† Location	**35,000 MWD/MTU 5-Year Cooling (mrem/hr) (mSv/h)**	**45,000 MWD/MTU 9-Year Cooling (mrem/hr) (mSv/h)**
1	297.76 (2.9776)	208.32 (2.0832)
2	0.02 (0.0002)	0.01 (0.0001)
3	127.41 (1.2741)	89.14 (0.8914)
4	51.49 (0.5149)	36.02 (0.3602)
5	0.72 (0.0072)	0.50 (0.0050)
6 (dry MPC)††	329.17 (3.2917)	230.30 (2.3030)
7 (no temp. shield)	1794.41 (17.9441)	1255.40 (12.5540)
7 (with temp. shield)	381.90 (3.8190)	267.18 (2.6718)

† Refer to Figure 5.1.1.

†† Overpack closure plate not included.

Table 5.4.7

DOSE RATES FROM NEUTRONS
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP
AND COOLING TIMES
(The values in parentheses are in mSv/h)

Dose Point† Location	**35,000 MWD/MTU 5-Year Cooling (mrem/hr) (mSv/h)**	**45,000 MWD/MTU 9-Year Cooling (mrem/hr) (mSv/h)**
1	65.63 (0.6563)	133.56 (1.3356)
2	19.40 (0.1940)	43.08 (0.4308)
3	29.88 (0.2988)	60.80 (0.6080)
4	17.76 (0.1776)	36.14 (0.3614)
5	26.45 (0.2645)	53.81 (0.5381)
6 (dry MPC)††	65.38 (0.6538)	133.03 (1.3303)
7 (no temp. shield)	325.90 (3.2590)	663.17 (6.6317)
7 (with temp. shield)	14.52 (0.1452)	29.55 (0.2955)

† Refer to Figure 5.1.1.

†† Overpack closure plate not included.

Table 5.4.8

TOTAL DOSE RATES
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP
AND COOLING TIMES
(The values in parentheses are in mSv/h)

Dose Point† Location	**40,000 MWD/MTU 5-Year Cooling (mrem/hr) (mSv/h)**	**47,500 MWD/MTU 8-Year Cooling (mrem/hr) (mSv/h)**
1	326.24 (3.2624)	316.06 (3.1606)
2	119.03 (1.1903)	90.69 (0.9069)
3	154.90 (1.5490)	177.40 (1.7740)
4	77.14 (0.7714)	92.12 (0.9212)
5	57.73 (0.5773)	92.43 (0.9243)
6 (dry MPC)††	439.28 (4.3928)	435.70 (4.3570)
7 (no temp. shield)	1929.94 (19.2994)	1782.95 (17.8295)
7 (with temp. shield)	377.94 (3.7794)	302.90 (3.0290)

† Refer to Figure 5.1.1.

†† Overpack closure plate not included.

Table 5.4.9

TOTAL DOSE RATES
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP AND COOLING TIMES
(The values in parentheses are in mSv/h)

Dose Point† Location	**35,000 MWD/MTU 5-Year Cooling (mrem/hr) (mSv/h)**	**45,000 MWD/MTU 9-Year Cooling (mrem/hr) (mSv/h)**
1	373.64 (3.7364)	347.35 (3.4735)
2	119.85 (1.1985)	95.35 (0.9535)
3	158.26 (1.5826)	150.75 (1.5075)
4	69.69 (0.6969)	72.52 (0.7252)
5	27.30 (0.2730)	54.52 (0.5452)
6 (dry MPC)††	403.87 (4.0387)	367.85 (3.6785)
7 (no temp. shield)	2184.76 (21.8476)	1949.86 (19.4986)
7 (with temp. shield)	416.78 (4.1678)	312.77 (3.1277)

† Refer to Figure 5.1.1.

†† Overpack closure plate not included.

Table 5.4.10

PEAK-TO-AVERAGE RATIOS FOR THE DOSE COMPONENTS AT VARIOUS LOCATIONS

Location	**Fuel Gammas**	**Gammas from Neutrons**	**^{60}Co Gammas**	**Neutron**
Pocket Trunnion (surface)	0.06	0.33	0.06	7.94
Steel Fin (surface)	0.74	0.95	0.74	2.16
Holtite-A (surface)	1.17	1.05	1.17	0.71
Pocket Trunnion (1 meter)	0.6	0.86	0.6	2.82
Steel Fin (1 meter)	1.0	1.0	1.0	1.05
Holtite-A (1 meter)	1.0	1.0	1.0	0.92

Table 5.4.11

DOSE RATES FOR NORMAL CONDITIONS SHOWING THE EFFECT OF PEAKING
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
40,000 MWD/MTU 5-YEAR COOLING
(The values in parentheses are in mSv/h)

Dose Point† Location	Fuel Gammas (mrem/hr) (mSv/h)	Gammas from Neutrons (mrem/hr) (mSv/h)	^{60}Co Gammas (mrem/hr) (mSv/h)	Neutrons (mrem/hr) (mSv/h)	Total (mrem/hr) (mSv/h)
SURFACE					
2	90.55 (0.9055)	6.33 (0.0633)	0.03 (0.0003)	22.12 (0.2212)	119.03 (1.1903)
2 (fin)	67.01 (0.6701)	6.01 (0.0601)	0.02 (0.0002)	47.78 (0.4778)	120.82 (1.2082)
2 (Holtite)	105.94 (1.0594)	6.65 (0.0665)	0.04 (0.0004)	15.71 (0.1571)	128.34 (1.2834)
2a (Holtite)††	12.35 (0.1235)	1.49 (0.0149)	77.43 (0.7743)	10.08 (0.1008)	101.35 (1.0135)
2a (pocket trunnion)††	0.63 (0.0063)	0.47 (0.0047)	3.97 (0.0397)	112.75 (1.1275)	117.82 (1.1782)
ONE METER					
2	40.47 (0.4047)	2.20 (0.0220)	1.06 (0.0106)	7.74 (0.0774)	51.47 (0.5147)
2 (fin)	40.47 (0.4047)	2.20 (0.0220)	1.06 (0.0106)	8.13 (0.0813)	51.86 (0.5186)
2 (Holtite)	40.47 (0.4047)	2.20 (0.0220)	1.06 (0.0106)	7.12 (0.0712)	50.85 (0.5085)
2a (Holtite)††	14.67 (0.1467)	0.93 (0.0093)	16.58 (0.1658)	6.90 (0.0690)	39.08 (0.3908)
2a (pocket trunnion)††	8.80 (0.0880)	0.80 (0.0080)	9.95 (0.0995)	21.14 (0.2114)	40.69 (0.4069)

† Refer to Figure 5.1.1.

†† Dose point #2a is axially located next to either the Holtite (neutron shield) or pocket trunnion and approximately 6 feet below Dose point #2.

Table 5.4.12

ANNUAL DOSE AT 300 METERS FROM A SINGLE CASK†
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
(The values in parentheses are in mSv/yr)

	40,000 MWD/MTU 5-Year Cooling (mrem/yr) (mSv/yr)	**47,500 MWD/MTU 8-Year Cooling (mrem/yr) (mSv/yr)**
Fuel gammas††	8.15 (0.0815)	4.34 (0.0434)
^{60}Co Gammas	2.46 (0.0246)	1.88 (0.0188)
Neutrons	2.94 (0.0294)	4.74 (0.0474)
Total	13.55 (0.1355)	10.96 (0.1096)

† 100% occupancy (8760 hours) is assumed.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.4.13

DOSE VALUES USED IN CALCULATING ANNUAL DOSE FROM
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
VARIOUS ISFSI CONFIGURATIONS
40,000 MWD/MTU AND 5-YEAR COOLING†
(The values in parentheses are in mSv/yr)

	A **Side of Overpack** **(mrem/yr)** **(mSv/yr)**	**B** **Top of Overpack** **(mrem/yr)** **(mSv/yr)**	**C** **Side of Shielded Overpack** **(mrem/yr)** **(mSv/yr)**
100 meters	337.58 (3.3758)	7.40 (0.0740)	110.31 (1.1031)
150 meters	115.93 (1.1593)	3.07 (0.0307)	40.56 (0.4056)
200 meters	51.52 (0.5152)	1.35 (0.0135)	17.54 (0.1754)
250 meters	24.53 (0.2453)	0.63 (0.0063)	8.81 (0.0881)
300 meters	13.28 (0.1328)	0.27 (0.0027)	4.15 (0.0415)
350 meters	6.76 (0.0676)	0.15 (0.0015)	2.23 (0.0223)
400 meters	3.28 (0.0328)	0.09 (0.0009)	1.16 (0.0116)

† 100% occupancy (8760 hours) is assumed.

Table 5.4.14

DOSE RATES AT THE CENTERLINE OF THE OVERPACK FOR DESIGN BASIS STAINLESS STEEL CLAD FUEL
(The values in parentheses are in mSv/h)

Dose Point† Location	**Fuel Gammas†† (mrem/hr) (mSv/h)**	**^{60}Co Gammas (mrem/hr) (mSv/h)**	**Neutrons (mrem/hr) (mSv/h)**	**Totals (mrem/hr) (mSv/h)**
MPC-24 (30,000 MWD/MTU AND 9-YEAR COOLING)				
2 (Adjacent)	101.34 (1.0134)	0.06 (0.0006)	5.13 (0.0513)	106.53 (1.0653)
2 (One Meter)	43.64 (0.4364)	0.40 (0.0040)	2.16 (0.0216)	46.20 (0.4620)
MPC-24 (40,000 MWD/MTU AND 15-YEAR COOLING)				
2 (Adjacent)	64.26 (0.6426)	0.01 (0.0001)	16.06 (0.1606)	80.33 (0.8033)
2 (One Meter)	28.38 (0.2838)	0.19 (0.0019)	5.63 (0.0563)	34.19 (0.3419)
MPC-68 (22,500 MWD/MTU AND 10-YEAR COOLING)				
2 (Adjacent)	80.71 (0.8071)	0.01 (0.0001)	1.51 (0.0151)	82.23 (0.8223)
2 (One Meter)	34.70 (0.3470)	0.30 (0.0030)	0.58 (0.0058)	35.59 (0.3559)

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.4.15

COMPARISON OF NEUTRON SOURCE PER INCH PER SECOND FOR DESIGN BASIS 7X7 FUEL AND DESIGN BASIS DRESDEN UNIT 1 FUEL
(The values in parentheses in second column are in cm, and The values in parentheses in third and fourth columns are in neutrons per sec per cm)

Assembly	**Active fuel length (inch) (cm)**	**Neutrons per sec per inch (neutrons per sec per cm)**	**Neutrons per sec per inch with Sb-Be source (neutrons per sec per cm)**	**Reference for neutrons per sec per inch**
7x7 design basis	144 (365.76)	5.60E+5 (2.20E+5)	N/A	Table 5.2.13 - 35 GWD/MTU and 5 year cooling
6x6 design basis	110 (279.4)	2.0e+5 (7.87E+4)	2.6E+5 (1.02E+5)	Table 5.2.14
6x6 design basis MOX	110 (279.4)	3.06E+5 (1.20E+5)	3.66E+5 (1.44E+5)	Table 5.2.17

Table 5.4.16

COMPARISON OF TOTAL DOSE RATES FOR DESIGN BASIS PWR FUEL
AND PWR FUEL WITH BPRAS
MPC-24 NORMAL AND ACCIDENT CONDITIONS
(The values in parentheses are in mSv/h)

Dose Point† Location	**40 GWD/MTU 5 year cooling DESIGN BASIS (mrem/hr) (mSv/h)**	**29 GWD/MTU 5 year cooling (mrem/hr) (mSv/h)**	**39 GWD/MTU 10 year cooling (mrem/hr) (mSv/h)**	**42.5 GWD/MTU 15 year cooling (mrem/hr) (mSv/h)**
BPRAs ?	NO	YES	YES	YES
SURFACE - NORMAL CONDITION				
1	326.24 (3.2624)	244.72 (2.4472)	195.95 (1.9595)	143.2 (1.4320)
2	119.03 (1.1903)	99.66 (0.9966)	69.57 (0.6957)	61.70 (0.6170)
3	154.90 (1.5490)	136.39 (1.3639)	136.84 (1.3684)	120.32 (1.2032)
4	77.14 (0.7714)	64.13 (0.6413)	67.66 (0.6766)	60.70 (0.6070)
5	57.73 (0.5773)	25.15 (0.2515)	48.25 (0.4825)	50.53 (0.5033)
6 (dry MPC)††	439.28 (4.3928)	445.46 (4.4546)	389.94 (3.8994)	324.45 (3.2445)
7 (no temp. shield)	1929.94 (19.2994)	1557.87 (15.5787)	1165.15 (11.6515)	810.77 (8.1077)
SURFACE - ACCIDENT CONDITION				
2	1371.34 (13.7134)	699.45 (6.9945)	1074.98 (10.7498)	1101.53 (11.0153)

† Refer to Figure 5.1.1.

†† Overpack closure plate not present.

Table 5.4.16 (continued)

COMPARISON OF TOTAL DOSE RATES FOR DESIGN BASIS PWR FUEL
AND PWR FUEL WITH BPRAS
MPC-24 NORMAL AND ACCIDENT CONDITIONS
(The values in parentheses are in mSv/h)

Dose Point† Location	**40 GWD/MTU 5 year cooling DESIGN BASIS (mrem/hr) (mSv/h)**	**29 GWD/MTU 5 year cooling (mrem/hr) (mSv/h)**	**39 GWD/MTU 10 year cooling (mrem/hr) (mSv/h)**	**42.5 GWD/MTU 15 year cooling (mrem/hr) (mSv/h)**
BPRAs ?	NO	YES	YES	YES
ONE METER - NORMAL CONDITION				
1	43.79 (0.4379)	34.47 (0.3447)	25.62 (0.2562)	19.12 (0.1912)
2	51.47 (0.5147)	44.42 (0.4442)	29.32 (0.2932)	25.50 (0.2550)
3	30.21 (0.3021)	28.68 (0.2868)	24.86 (0.2486)	21.27 (0.2127)
4	28.61 (0.2861)	27.06 (0.2706)	24.19 (0.2419)	20.68 (0.2068)
5	17.11 (0.1711)	7.55 (0.0755)	14.3 (0.1430)	14.95 (0.1495)
7 (no temp. shield)	889.68 (8.8968)	717.64 (7.1764)	501.61 (5.0161)	329.5 (3.2950)
ONE METER - ACCIDENT CONDITION				
2	491.73 (4.9173)	263.82 (2.6382)	378.91 (3.7891)	384.75 (3.8475)

† Refer to Figure 5.1.1.

Table 5.4.17

COMPARISON OF TOTAL DOSE RATES FOR DESIGN BASIS PWR FUEL
AND PWR FUEL WITH TPDS
MPC-24 NORMAL CONDITIONS
(The values in parentheses are in mSv/h)

Dose Point† Location	**40 GWD/MTU 5 year cooling DESIGN BASIS (mrem/hr) (mSv/h)**	**29 GWD/MTU 5 year cooling (mrem/hr) (mSv/h)**	**39 GWD/MTU 10 year cooling (mrem/hr) (mSv/h)**	**42.5 GWD/MTU 15 year cooling (mrem/hr) (mSv/h)**
TPDs ?	NO	YES	YES	YES
SURFACE - NORMAL CONDITION				
1	326.24 (3.2624)	241.88 (2.4188)	193.11 (1.9311)	140.36 (1.4036)
2	119.03 (1.1903)	78.00 (0.7800)	47.91 (0.4791)	40.04 (0.4004)
3	154.90 (1.5490)	134.01 (1.3401)	134.47 (1.3447)	117.95 (1.1795)
4	77.14 (0.7714)	63.09 (0.6309)	66.62 (0.6662)	59.66 (0.5966)
5	57.73 (0.5773)	25.12 (0.2512)	48.22 (0.4822)	50.51 (0.5051)
6 (dry MPC)††	439.28 (4.3928)	439.39 (4.3939)	383.86 (3.8386)	318.38 (3.1838)
7 (no temp. shield)	1929.94 (19.2994)	1531.39 (15.3139)	1138.67 (11.3867)	784.28 (7.8428)

† Refer to Figure 5.1.1.

†† Overpack closure plate not present.

Table 5.4.17 (continued)

COMPARISON OF TOTAL DOSE RATES FOR DESIGN BASIS PWR FUEL
AND PWR FUEL WITH TPDS
MPC-24 NORMAL CONDITIONS
(The values in parentheses are in mSv/h)

Dose Point† Location	**40 GWD/MTU 5 year cooling DESIGN BASIS (mrem/hr) (mSv/h)**	**29 GWD/MTU 5 year cooling (mrem/hr) (mSv/h)**	**39 GWD/MTU 10 year cooling (mrem/hr) (mSv/h)**	**42.5 GWD/MTU 15 year cooling (mrem/hr) (mSv/h)**
TPDs ?	NO	YES	YES	YES
ONE METER - NORMAL CONDITION				
1	43.79 (0.4379)	32.23 (0.3223)	23.39 (0.2339)	16.88 (0.1688)
2	51.47 (0.5147)	34.70 (0.3470)	19.61 (0.1961)	15.78 (0.1578)
3	30.21 (0.3021)	27.55 (0.2755)	23.74 (0.2374)	20.15 (0.2015)
4	28.61 (0.2861)	26.11 (0.2611)	23.25 (0.2325)	19.73 (0.1973)
5	17.11 (0.1711)	7.54 (0.0754)	14.29 (0.1429)	14.94 (0.1494)
7 (no temp. shield)	889.68 (8.8968)	703.89 (7.0389)	487.86 (4.8786)	315.76 (3.1576)

† Refer to Figure 5.1.1.

Table 5.4.18

DOSE RATES FROM FUEL GAMMAS
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP
AND COOLING TIMES†
(The values in parentheses are in mSv/h)

Dose Point†† Location	**40,000 MWD/MTU 8-Year Cooling (mrem/hr) (mSv/h)**	**45,000 MWD/MTU 11-Year Cooling (mrem/hr) (mSv/h)**
1	5.59 (0.0559)	4.33 (0.0433)
2	49.16 (0.4916)	38.88 (0.3888)
3	1.95 (0.0195)	1.66 (0.0166)
4	0.97 (0.0097)	0.81 (0.0081)
5	0.25 (0.0025)	0.3 (0.003)
6 (dry MPC)†††	12.96 (0.1296)	9.65 (0.0965)
7 (no temp. shield)	64.04 (0.6404)	45.37 (0.4537)
7 (with temp. shield)	20.4 (0.204)	17.75 (0.1775)

† Gammas generated by neutron capture are included with fuel gammas.

†† Refer to Figure 5.1.1.

††† Overpack closure plate not present.

Table 5.4.19

DOSE RATES FROM ^{60}Co GAMMAS
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP
AND COOLING TIMES
(The values in parentheses are in mSv/h)

Dose Point† Location	**40,000 MWD/MTU 8-Year Cooling (mrem/hr) (mSv/h)**	**45,000 MWD/MTU 11-Year Cooling (mrem/hr) (mSv/h)**
1	220.07 (2.2007)	159.87 (1.5987)
2	0.03 (0.0003)	0.02 (0.0002)
3	80.91 (0.8091)	58.78 (0.5878)
4	35.84 (0.3584)	26.04 (0.2604)
5	0.58 (0.0058)	0.42 (0.0042)
6 (dry MPC)††	258.78 (2.5878)	187.99 (1.8799)
7 (no temp. shield)	1476.27 (14.7627)	1072.43 (10.7243)
7 (with temp. shield)	323.69 (3.2369)	235.15 (2.3515)

† Refer to Figure 5.1.1.

†† Overpack closure plate not present.

Table 5.4.20

DOSE RATES FROM NEUTRONS
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP AND COOLING TIMES
(The values in parentheses are in mSv/h)

Dose Point† Location	**40,000 MWD/MTU 8-Year Cooling (mrem/hr) (mSv/h)**	**45,000 MWD/MTU 11-Year Cooling (mrem/hr) (mSv/h)**
1	84.52 (0.8452)	110.33 (1.1033)
2	23.37 (0.2337)	30.51 (0.3051)
3	67.31 (0.6731)	87.86 (0.8786)
4	39.15 (0.3915)	51.1 (0.511)
5	58.78 (0.5878)	76.73 (0.7673)
6 (dry MPC)††	141.89 (1.4189)	185.22 (1.8522)
7 (no temp. shield)	438.45 (4.3845)	572.34 (5.7234)
7 (with temp. shield)	20.57 (0.2057)	26.85 (0.2685)

† Refer to Figure 5.1.1.

†† Overpack closure plate not included.

Table 5.4.21

TOTAL DOSE RATES
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP
AND COOLING TIMES
(The values in parentheses are in mSv/h)

Dose Point† Location	**40,000 MWD/MTU 8-Year Cooling (mrem/hr) (mSv/h)**	**45,000 MWD/MTU 11-Year Cooling (mrem/hr) (mSv/h)**
1	310.18 (3.1018)	274.53 (2.7453)
2	72.56 (0.7256)	69.41 (0.6941)
3	150.16 (1.5016)	148.3 (1.483)
4	75.96 (0.7596)	77.95 (0.7795)
5	59.6 (0.596)	77.45 (0.7745)
6 (dry MPC)††	413.62 (4.1362)	382.85 (3.8285)
7 (no temp. shield)	1978.75 (19.7875)	1690.14 (16.9014)
7 (with temp. shield)	364.66 (3.6466)	279.74 (2.7974)

† Refer to Figure 5.1.1.

†† Overpack closure plate not included.

Table 5.4.22

ANNUAL DOSE AT 300 METERS FROM A SINGLE CASK†
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL
(The values in parentheses are in mSv/yr)

	40,000 MWD/MTU 8-Year Cooling (mrem/yr) (mSv/yr)	**45,000 MWD/MTU 11-Year Cooling (mrem/yr) (mSv/yr)**
Fuel gammas††	3.83 (0.0383)	3.04 (0.0304)
^{60}Co Gammas	2.25 (0.0225)	1.63 (0.0163)
Neutrons	3.08 (0.0308)	4.03 (0.0403)
Total	9.16 (0.0916)	8.7 (0.087)

† 100% occupancy (8760 hours) is assumed.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.4.23

DOSE VALUES USED IN CALCULATING ANNUAL DOSE FROM
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL
VARIOUS ISFSI CONFIGURATIONS
40,000 MWD/MTU AND 8-YEAR COOLING†
(The values in parentheses are in mSv/yr)

	A **Side of Overpack (mrem/yr)** **(mSv/yr)**	**B** **Top of Overpack (mrem/yr)** **(mSv/yr)**	**C** **Side of Shielded Overpack (mrem/yr)** **(mSv/yr)**
100 meters	234.04 (2.3404)	7.24 (0.0724)	82.67 (0.8267)
150 meters	81.85 (0.8185)	2.94 (0.0294)	30 (0.3)
200 meters	35.65 (0.3565)	1.31 (0.0131)	13.29 (0.1329)
250 meters	17.27 (0.1727)	0.63 (0.0063)	6.37 (0.0637)
300 meters	8.84 (0.0884)	0.32 (0.0032)	3.38 (0.0338)
350 meters	4.85 (0.0485)	0.17 (0.0017)	1.79 (0.0179)
400 meters	2.63 (0.0263)	0.07 (0.0007)	0.97 (0.0097)

† 100% occupancy (8760 hours) is assumed.

Table 5.4.24

COMPARISON OF TOTAL DOSE RATES FOR DESIGN BASIS PWR FUEL
TO PWR FUEL REPRESENTATIVE BURNUP-COOLING TIME COMBINATIONS
MPC-32 NORMAL AND ACCIDENT CONDITIONS
(The values in parentheses are in mSv/h)

Dose Point† Location	**40 GWD/MTU 8 year cooling DESIGN BASIS (mrem/hr) (mSv/h)**	**25 GWD/MTU 8 year cooling (mrem/hr) (mSv/h)**	**45 GWD/MTU 20 year cooling (mrem/hr) (mSv/h)**	**25 GWD/MTU 12 year cooling (mrem/hr) (mSv/h)**	**35 GWD/MTU 16 year cooling (mrem/hr) (mSv/h)**	**43 GWD/MTU 20 year cooling (mrem/hr) (mSv/h)**
Non-zircaloy Grid Spacer ?	No	No	No	Yes	Yes	Yes
SURFACE - NORMAL CONDITION						
1	310.18 (3.1018)	186.77 (1.8677)	130.26 (1.3026)	117.58 (1.1758)	114.3 (1.143)	123.81 (1.2381)
2	72.56 (0.7256)	29.16 (0.2916)	40.8 (0.408)	35.37 (0.3537)	39.99 (0.3999)	46.67 (0.4667)
3	150.16 (1.5016)	76.85 (0.7685)	81.95 (0.8195)	50.26 (0.5026)	59.48 (0.5948)	76.7 (0.767)
4	75.96 (0.7596)	36.24 (0.3624)	45.08 (0.4508)	24.19 (0.2419)	30.93 (0.3093)	42.05 (0.4205)
5	59.6 (0.596)	13.97 (0.1397)	55.39 (0.5539)	12.03 (0.1203)	28.77 (0.2877)	50.84 (0.5084)
7 (no temp. shield)	1978.75 (19.7875)	1238.1 (12.381)	758.4 (7.584)	782.41 (7.8241)	719.89 (7.1989)	730.86 (7.3086)
SURFACE - ACCIDENT CONDITION						
2	1355.35 (13.5535)	341.13 (3.4113)	1201.76 (12.0176)	320.62 (3.2062)	667.1 (6.671)	1125.03 (11.2503)

† Refer to Figure 5.1.1.

Table 5.4.24 (continued)

COMPARISON OF TOTAL DOSE RATES FOR DESIGN BASIS PWR FUEL
TO PWR FUEL REPRESENTATIVE BURNUP-COOLING TIME COMBINATIONS
MPC-32 NORMAL AND ACCIDENT CONDITIONS
(The values in parentheses are in mSv/h)

Dose Point† Location	**40 GWD/MTU 8 year cooling DESIGN BASIS (mrem/hr) (mSv/h)**	**25 GWD/MTU 8 year cooling (mrem/hr) (mSv/h)**	**45 GWD/MTU 20 year cooling (mrem/hr) (mSv/h)**	**25 GWD/MTU 12 year cooling (mrem/hr) (mSv/h)**	**35 GWD/MTU 16 year cooling (mrem/hr) (mSv/h)**	**43 GWD/MTU 20 year cooling (mrem/hr) (mSv/h)**
Non-zircaloy Grid Spacer ?	No	No	No	Yes	Yes	Yes
ONE METER - NORMAL CONDITION						
1	38.14 (0.3814)	22.4 (0.2240)	15.94 (0.1594)	15.53 (0.1553)	15.03 (0.1503)	15.92 (0.1592)
2	30.67 (0.3067)	13.04 (0.1304)	16.12 (0.1612)	15.62 (0.1562)	16.92 (0.1692)	18.93 (0.1893)
3	25.75 (0.2575)	13.77 (0.1377)	12.58 (0.1258)	9.87 (0.0987)	10.58 (0.1058)	12.4 (0.124)
4	25.78 (0.2578)	14.02 (0.1402)	12.55 (0.1255)	9.58 (0.0958)	10.28 (0.1028)	12.13 (0.1213)
5	17.1 (0.171)	4.07 (0.0407)	15.8 (0.1580)	3.48 (0.0348)	8.24 (0.0824)	14.51 (0.1451)
7 (no temp. shield)	863.73 (8.6373)	576.93 (5.7693)	280.28 (2.8028)	360.43 (3.6043)	303.86 (3.0386)	274.44 (2.7444)
ONE METER - ACCIDENT CONDITION						
2	471.11 (4.7111)	122.5 (1.2250)	409.61 (4.0961)	116.96 (1.1696)	232.74 (2.3274)	385.95 (3.8595)

† Refer to Figure 5.1.1.

Table 5.4.25

DESCRIPTION OF ANALYZED MPC-32 PWR FUEL NON-ZIRCALOY GRID SPACER

Description	Value
Inconel incore Grid Spacers (kg)	4.9
Co-59 Impurity Level (g/kg)	4.7

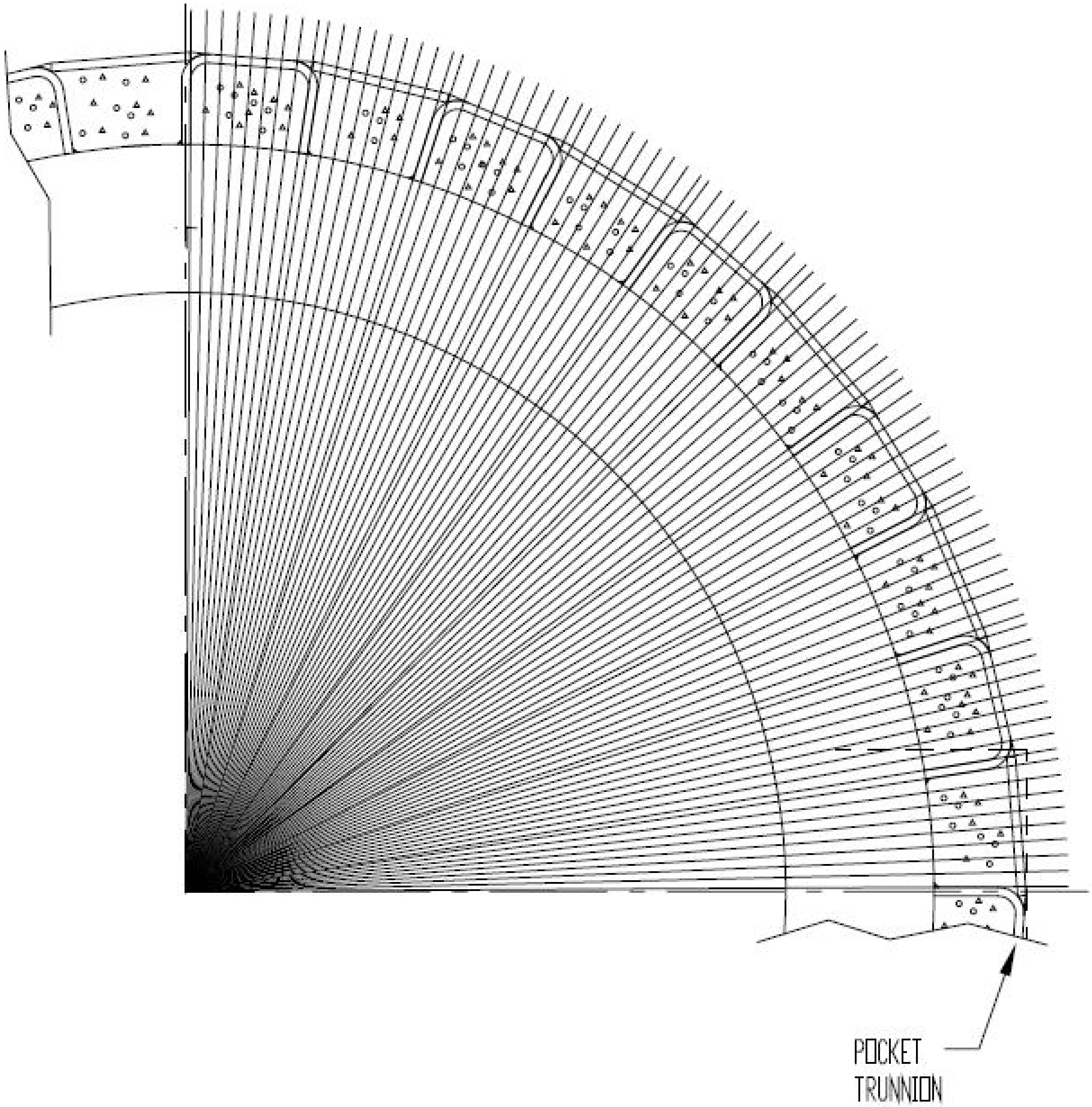


FIGURE 5.4.1; DEPICTION OF THE AZIMUTHAL SEGMENTATION OF THE OVERPACK USED IN ANALYZING NEUTRON AND PHOTON STREAMING

5.5 REGULATORY COMPLIANCE

Chapters 1 and 2 and this chapter of this FSAR describe in detail the shielding structures, systems, and components (SSCs) important to safety.

This chapter has evaluated these shielding SSCs important to safety and has assessed the impact on health and safety resulting from operation of an independent spent fuel storage installation (ISFSI) utilizing the HI-STAR 100 System.

It has been shown that the design of the shielding system of the HI-STAR 100 System is in compliance with 10CFR72 and that the applicable design and acceptance criteria including 10CFR20 have been satisfied. Thus, this shielding evaluation provides reasonable assurance that the HI-STAR 100 System will allow safe storage of spent fuel.

5.6 REFERENCES

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[5.4.3] D. J. Whalen, et al., "MCNP: Neutron Benchmark Problems," LA-12212, Los Alamos National Laboratory, November 1991.

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[5.4.5] S. E. Turner, "Uncertainty Analysis - Axial Burnup Distribution Effects," presented in "Proceedings of a Workshop on the Use of Burnup Credit in Spent Fuel Transport Casks," SAND-89-0018, Sandia National Laboratory, Oct. 1989.

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APPENDIX 5.A

SAMPLE INPUT FILE FOR SAS2H

(Total number of pages in this appendix : 4)

```
=SAS2H    PARM='halt08,skipshipdata'
bw 15x15 PWR assembly
' fuel temp 923
44groupndf5    LATTICECELL
UO2 1 0.95 923 92234 0.03026 92235 3.4 92236 0.01564
    92238 96.5541 END
'
' Zirc 4 composition
ARBM-ZIRC4 6.55 4 1 0 0 50000 1.7 26000 0.24 24000 0.13 40000 97.93
           2 1.0 595 END
'
' water with 652.5 ppm boron
H2O       3 DEN=0.7135 1  579 END
ARBM-BORMOD  0.7135 1 1 0 0 5000 100 3 652.5E-6  579  END
'
co-59  3 0 1-20 579 end
kr-83  1 0 1-20 923 end
kr-84  1 0 1-20 923 end
kr-85  1 0 1-20 923 end
kr-86  1 0 1-20 923 end
sr-90  1 0 1-20 923 end
y-89   1 0 1-20 923 end
zr-94  1 0 1-20 923 end
zr-95  1 0 1-20 923 end
mo-94  1 0 1-20 923 end
mo-95  1 0 1-20 923 end
nb-94  1 0 1-20 923 end
nb-95  1 0 1-20 923 end
tc-99  1 0 1-20 923 end
ru-106  1 0 1-20 923 end
rh-103  1 0 1-20 923 end
rh-105  1 0 1-20 923 end
sb-124  1 0 1-20 923 end
sn-126  1 0 1-20 923 end
xe-131  1 0 1-20 923 end
xe-132  1 0 1-20 923 end
xe-134  1 0 1-20 923 end
'
xe-135  1 0 1-09 923 end
'
xe-136  1 0 1-20 923 end
cs-133  1 0 1-20 923 end
cs-134  1 0 1-20 923 end
cs-135  1 0 1-20 923 end
cs-137  1 0 1-20 923 end
ba-136  1 0 1-20 923 end
la-139  1 0 1-20 923 end
ce-144  1 0 1-20 923 end
pr-143  1 0 1-20 923 end
nd-143  1 0 1-20 923 end
nd-144  1 0 1-20 923 end
nd-145  1 0 1-20 923 end
nd-146  1 0 1-20 923 end
nd-147  1 0 1-20 923 end
nd-148  1 0 1-20 923 end
```

```
nd-150  1 0 1-20 923 end
pm-147  1 0 1-20 923 end
pm-148  1 0 1-20 923 end
pm-149  1 0 1-20 923 end
sm-147  1 0 1-20 923 end
sm-148  1 0 1-20 923 end
sm-149  1 0 1-20 923 end
sm-150  1 0 1-20 923 end
sm-151  1 0 1-20 923 end
sm-152  1 0 1-20 923 end
eu-151  1 0 1-20 923 end
eu-153  1 0 1-20 923 end
eu-154  1 0 1-20 923 end
eu-155  1 0 1-20 923 end
gd-154  1 0 1-20 923 end
gd-155  1 0 1-20 923 end
gd-157  1 0 1-20 923 end
gd-158  1 0 1-20 923 end
gd-160  1 0 1-20 923 end
'
END COMP
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'  FUEL-PIN-CELL GEOMETRY:
'
SQUAREPITCH  1.44272 0.950468 1 3 1.08712 2 0.97028 0 END
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'  MTU in this model is 0.495485 based on fuel dimensions provided
'
'  1 power cycle will be used and a library will be generated every
'  2500 MWD/MTU  power level is 40 MW/MTU
'  therefore 62.5 days per 2500 MWD/MTU
'  Below
'  BURN=62.5*NLIB/CYC
'  POWER=MTU*40
'
'  Number of libraries is 17 which is 42,500 MWD/MTU burnup (17*2500)
'
'  ASSEMBLY AND CYCLE PARAMETERS:
'
NPIN/ASSM=208 FUELNGTH=365.76 NCYCLES=1 NLIB/CYC=17
PRINTLEVEL=1
LIGHTEL=5  INPLEVEL=1   NUMHOLES=17
NUMINStr= 0  ORTUBE= 0.6731  SRTUBE=0.63246   END
POWER=19.81938 BURN=1062.5 END
'
 O  66.54421
 FE  0.24240868
 ZR 98.78151  CR  0.1311304   SN  1.714782
'
END
=SAS2H    PARM='restarts,halt17,skipshipdata'
```

```
bw 15x15 PWR assembly
END
```

APPENDIX 5.B

SAMPLE INPUT FILE FOR ORIGEN-S

(Total number of pages in this appendix : 7)

```
#ORIGENS
0$$  A4 33 A8 26 A11 71  E
1$$  1 T
bw 15x15 FUEL -- FT33F001 -
'
' SUBCASE  1 LIBRARY POSITION  1
'
'   lib     pos      grms   photon group
3$$ 33  A3  1  0  A16 2   E T
35$$ 0 T
56$$  5  5  A6 3  A10  0  A13 9  A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.0625 E T
 FUEL 3.4
BW 15x15 0.495485 MTU
58** 19.81938  19.81938 19.81938 19.81938  19.81938
60**  1.0000   3.0000  15.0000  30.0000  62.5
66$$ A1 2 A5 2 A9 2  E
73$$  922350  922340  922360  922380  80000  500000
      260000  240000  400000
74** 16846.48 149.9336 77.49379 478410.7 66544.21 1714.782
     242.0868 131.1304 98781.51
75$$  2  2  2  2  4  4  4  4  4  T
'
' SUBCASE  2 LIBRARY POSITION  2
'
3$$ 33  A3   2  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   5  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE  3 LIBRARY POSITION  3
'
3$$ 33  A3   3  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE  4 LIBRARY POSITION  4
'
3$$ 33  A3   4  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
```

```
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE  5 LIBRARY POSITION  5
'
3$$ 33  A3   5  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE  6 LIBRARY POSITION  6
'
3$$ 33  A3   6  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE  7 LIBRARY POSITION  7
'
3$$ 33  A3   7  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE  8 LIBRARY POSITION  8
'
3$$ 33  A3   8  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE  9 LIBRARY POSITION  9
'
3$$ 33  A3   9  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
```

```
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE 10 LIBRARY POSITION 10
'
3$$ 33  A3  10  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE 11 LIBRARY POSITION 11
'
3$$ 33  A3  11  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE 12 LIBRARY POSITION 12
'
3$$ 33  A3  12  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE 13 LIBRARY POSITION 13
'
3$$ 33  A3  13  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE 14 LIBRARY POSITION 14
```

```
'
3$$ 33  A3  14  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE 15 LIBRARY POSITION 15
'
3$$ 33  A3  15  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE 16 LIBRARY POSITION 16
'
3$$ 33  A3  16  A4 7 0  A16 2  A33 18  E T
35$$ 0 T
56$$  3  3  A6 1  A10  3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE - decay
'
54$$ A8 1 E
56$$ 0 9 A6 1 A10 3 A14 3 A15 1 A19 1 E
57** 0.0 0 1.E-5  E T
 fuel enrichment above
60** 0.5 0.75 1.0 4.0 8.0 12.0 24.0 48.0 96.0
61** F0.1
65$$
'GRAM-ATOMS   GRAMS    CURIES     WATTS-ALL  WATTS-GAMMA
     3Z     0  1  0     0 0 0      1 0 0         3Z         6Z
     3Z     0  1  0     0 0 0      1 0 0         3Z         6Z
     3Z     0  1  0     0 0 0      1 0 0         3Z         6Z  T
'
' SUBCASE - decay
'
54$$ A8 1 E
56$$ 0 9 A6 1 A10 9 A14 4 A15 1 A19 1 E
57** 4.0 0 1.E-5  E T
 fuel enrichment above
60** 10.0 20.0 30.0 60.0 90.0 120.0 180.0 240.0 365.0
```

```
61** F0.1
65$$
'GRAM-ATOMS   GRAMS    CURIES     WATTS-ALL  WATTS-GAMMA
     3Z     0  1  0     0 0 0      1 0 0         3Z         6Z
     3Z     0  1  0     0 0 0      1 0 0         3Z         6Z
     3Z     0  1  0     0 0 0      1 0 0         3Z         6Z  T
'
' SUBCASE - decay
'
54$$ A8 0 E
56$$ 0 9 A6 1 A10 9 A14 5 A15 1 A19 1 E
57** 1.0 0 1.E-5  E T
 fuel enrichment above
60** 1.5 3.0 4.0 5.0 6.0 7.0 8.0 9.0 10.0
61** F1.0e-5
65$$
'GRAM-ATOMS   GRAMS    CURIES     WATTS-ALL  WATTS-GAMMA
     3Z     0  1  0     1 0 0      1 0 0         3Z         6Z
     3Z     0  1  0     1 0 0      1 0 0         3Z         6Z
     3Z     0  1  0     1 0 0      1 0 0         3Z         6Z
81$$ 2 0 26 1 E
82$$ 0 2 2 2 2 2 2 2 2
83**  1.1E+7  8.0E+6 6.0E+6 4.0E+6 3.0E+6 2.5E+6 2.0E+6 1.5E+6
      1.0E+6  7.0E+5 4.5E+5 3.0E+5 1.5E+5 1.0E+5 7.0E+4 4.5E+4
      3.0E+4  2.0E+4 1.0E+4
84**  20.0E+6 6.43E+6 3.0E+6 1.85E+6 1.40E+6 9.00E+5 4.00E+5 1.0E+5 T
'
'
'
'
'
'
'
'
' SUBCASE - decay
'
54$$ A8 0 E
56$$ 0 10 A6 1 A10 9 A14 5 A15 1 A19 1 E
57** 10.0 0 1.E-5  E T
 fuel enrichment above
60** 11.0 12.0 13.0 14.0 15.0 16.0 17.0 18.0 19.0 20.0
61** F1.0e-5
65$$
'GRAM-ATOMS   GRAMS    CURIES     WATTS-ALL  WATTS-GAMMA
     3Z     0  1  0     1 0 0      1 0 0         3Z         6Z
     3Z     0  1  0     1 0 0      1 0 0         3Z         6Z
     3Z     0  1  0     1 0 0      1 0 0         3Z         6Z
81$$ 2 0 26 1 E
82$$ 2 2 2 2 2 2 2 2 2 2
83**  1.1E+7  8.0E+6 6.0E+6 4.0E+6 3.0E+6 2.5E+6 2.0E+6 1.5E+6
      1.0E+6  7.0E+5 4.5E+5 3.0E+5 1.5E+5 1.0E+5 7.0E+4 4.5E+4
      3.0E+4  2.0E+4 1.0E+4
84**  20.0E+6 6.43E+6 3.0E+6 1.85E+6 1.40E+6 9.00E+5 4.00E+5 1.0E+5 T
'
'
'
```

```
'
'
'
'
'
'
'
' END
'
56$$ F0 T
END
```

APPENDIX 5.C

SAMPLE INPUT FILE FOR MCNP

(Total number of pages in this appendix : 39)

```
message:   outp=m68n65ao  srctp=m68n65as  runtpe=m68n65ar
           mctal=m68n65am  wssa=m68n65aw  rssa=pt001w

m68n65a
c       HI STAR 100 MPC68
c
c
c     two pocket trunions modeled
c     holtite present
c     impact limiters not present
c     axial model
c
c     origen is at the bottom of the overpack - as an example of the origen
c     item 1 on drawing 1397 is 6 inches thick and extends
c     from 0.0 to 6.0 inches in the axial direction
c
c     only cells that contain material are split axially
c     importance splitting is not done in cells with 0 material
c
c     universe 1
c
c      egg crate
301    0           -30            -400 u=1
302    0            31            -400 u=1
303    0        30 -31 -32        -400 u=1
304    0        30 -31  33        -400 u=1
305    5 -7.92         28     400 -420 u=1
306    5 -7.92            -23 400 -420 u=1
307    5 -7.92     -15 23 -28 400 -420 u=1
308    5 -7.92  20     23 -28 400 -420 u=1
309    5 -7.92         28     420 -430 u=1
310    5 -7.92            -23 420 -430 u=1
311    5 -7.92     -15 23 -28 420 -430 u=1
312    5 -7.92  20     23 -28 420 -430 u=1
313    5 -7.92         28     430 -440 u=1
314    5 -7.92            -23 430 -440 u=1
315    5 -7.92     -15 23 -28 430 -440 u=1
316    5 -7.92  20     23 -28 430 -440 u=1
317    5 -7.92         28     440 -670 u=1
318    5 -7.92            -23 440 -670 u=1
319    5 -7.92     -15 23 -28 440 -670 u=1
320    5 -7.92  20     23 -28 440 -670 u=1
321    5 -7.92         28     670 -460 u=1
322    5 -7.92            -23 670 -460 u=1
323    5 -7.92     -15 23 -28 670 -460 u=1
324    5 -7.92  20     23 -28 670 -460 u=1
c      boral and inside of egg crate and outside of fuel
326    0        15 -30 23 -28 400 -410 u=-1
327    0        31 -20 23 -28 400 -410 u=-1
328    0        30 -31 33 -28 400 -410 u=-1
329    0        30 -31 23 -32 400 -410 u=-1
330    0        15 -18 26 -28 410 -435 u=-1
331    0        19 -20 26 -28 410 -435 u=-1
332    0        15 -17 23 -24 410 -435 u=-1
333    0        15 -17 25 -26 410 -435 u=-1
334    6 -2.644 18 -19 27 -28 410 -420 u=-1
335    5 -7.92  18 -19 26 -27 410 -420 u=-1
336    6 -2.644 15 -16 24 -25 410 -420 u=-1
337    5 -7.92  16 -17 24 -25 410 -420 u=-1
338    6 -2.644 18 -19 27 -28 420 -430 u=-1
339    5 -7.92  18 -19 26 -27 420 -430 u=-1
```

```
340    6 -2.644 15 -16 24 -25 420 -430 u=-1
341    5 -7.92  16 -17 24 -25 420 -430 u=-1
342    6 -2.644 18 -19 27 -28 430 -435 u=-1
343    5 -7.92  18 -19 26 -27 430 -435 u=-1
344    6 -2.644 15 -16 24 -25 430 -435 u=-1
345    5 -7.92  16 -17 24 -25 430 -435 u=-1
346    0        17 -30 23 -26 410 -435 u=-1
347    0        31 -20 23 -26 410 -435 u=-1
348    0        30 -31 23 -32 410 -435 u=-1
349    0        30 -31 33 -26 410 -435 u=-1
350    0        15 -30 23 -28 435 -460 u=-1
351    0        31 -20 23 -28 435 -460 u=-1
352    0        30 -31 33 -28 435 -460 u=-1
353    0        30 -31 23 -32 435 -460 u=-1
c      fuel element
354    5 -1.51304  30 -31 32 -33     -420 u=1
355    2 -3.979996 30 -31 32 -33 420 -425 u=-1
356    0           30 -31 32 -33 425 -430 u=-1
357    5 -0.270112 30 -31 32 -33 430 -440 u=-1
358    5 -0.689740 30 -31 32 -33 440 -445 u=-1
359    5 -1.393935 30 -31 32 -33 445 -450 u=-1
360    5 -0.261853 30 -31 32 -33 450 -670 u=-1
361    5 -0.261853 30 -31 32 -33 670 -455 u=-1
362    0           30 -31 32 -33 455  u=1
363    0           -30           460 u=1
364    0            31           460 u=1
365    0        30 -31 -32       460 u=1
366    0        30 -31  33       460 u=1
c
c     universe 2
c
c      egg crate
401    0           -30           -400 u=2
402    0            31           -400 u=2
403    0        30 -31 -32       -400 u=2
404    0        30 -31  33       -400 u=2
405    5 -7.92         28     400 -420 u=2
406    5 -7.92            -23 400 -420 u=2
407    5 -7.92     -15 23 -28 400 -420 u=2
408    5 -7.92  20     23 -28 400 -420 u=2
409    5 -7.92         28     420 -430 u=2
410    5 -7.92            -23 420 -430 u=2
411    5 -7.92     -15 23 -28 420 -430 u=2
412    5 -7.92  20     23 -28 420 -430 u=2
413    5 -7.92         28     430 -440 u=2
414    5 -7.92            -23 430 -440 u=2
415    5 -7.92     -15 23 -28 430 -440 u=2
416    5 -7.92  20     23 -28 430 -440 u=2
417    5 -7.92         28     440 -670 u=2
418    5 -7.92            -23 440 -670 u=2
419    5 -7.92     -15 23 -28 440 -670 u=2
420    5 -7.92  20     23 -28 440 -670 u=2
421    5 -7.92         28     670 -460 u=2
422    5 -7.92            -23 670 -460 u=2
423    5 -7.92     -15 23 -28 670 -460 u=2
424    5 -7.92  20     23 -28 670 -460 u=2
c      boral and inside of egg crate and outside of fuel
426    0        15 -30 23 -28 400 -460 u=-2
427    0        31 -20 23 -28 400 -460 u=-2
428    0        30 -31 33 -28 400 -460 u=-2
429    0        30 -31 23 -32 400 -460 u=-2
```

```
c      fuel element
454    5 -1.51304  30 -31 32 -33     -420 u=2
455    2 -3.979996 30 -31 32 -33 420 -425 u=-2
456    0           30 -31 32 -33 425 -430 u=-2
457    5 -0.270112 30 -31 32 -33 430 -440 u=-2
458    5 -0.689740 30 -31 32 -33 440 -445 u=-2
459    5 -1.393935 30 -31 32 -33 445 -450 u=-2
460    5 -0.261853 30 -31 32 -33 450 -670 u=-2
461    5 -0.261853 30 -31 32 -33 670 -455 u=-2
462    0           30 -31 32 -33 455  u=2
463    0           -30           460 u=2
464    0            31           460 u=2
465    0        30 -31 -32       460 u=2
466    0        30 -31  33       460 u=2
c
c     universe 3
c
c      egg crate
501    0           -30           -400 u=3
502    0            31           -400 u=3
503    0        30 -31 -32       -400 u=3
504    0        30 -31  33       -400 u=3
505    5 -7.92         28     400 -420 u=3
506    5 -7.92            -23 400 -420 u=3
507    5 -7.92     -15 23 -28 400 -420 u=3
508    5 -7.92  20     23 -28 400 -420 u=3
509    5 -7.92         28     420 -430 u=3
510    5 -7.92            -23 420 -430 u=3
511    5 -7.92     -15 23 -28 420 -430 u=3
512    5 -7.92  20     23 -28 420 -430 u=3
513    5 -7.92         28     430 -440 u=3
514    5 -7.92            -23 430 -440 u=3
515    5 -7.92     -15 23 -28 430 -440 u=3
516    5 -7.92  20     23 -28 430 -440 u=3
517    5 -7.92         28     440 -670 u=3
518    5 -7.92            -23 440 -670 u=3
519    5 -7.92     -15 23 -28 440 -670 u=3
520    5 -7.92  20     23 -28 440 -670 u=3
521    5 -7.92         28     670 -460 u=3
522    5 -7.92            -23 670 -460 u=3
523    5 -7.92     -15 23 -28 670 -460 u=3
524    5 -7.92  20     23 -28 670 -460 u=3
c      boral and inside of egg crate and outside of fuel
526    0        15 -30 23 -28 400 -410 u=-3
527    0        31 -20 23 -28 400 -410 u=-3
528    0        30 -31 33 -28 400 -410 u=-3
529    0        30 -31 23 -32 400 -410 u=-3
530    0        15 -18 26 -28 410 -435 u=-3
531    0        19 -20 26 -28 410 -435 u=-3
534    6 -2.644 18 -19 27 -28 410 -420 u=-3
535    5 -7.92  18 -19 26 -27 410 -420 u=-3
538    6 -2.644 18 -19 27 -28 420 -430 u=-3
539    5 -7.92  18 -19 26 -27 420 -430 u=-3
542    6 -2.644 18 -19 27 -28 430 -435 u=-3
543    5 -7.92  18 -19 26 -27 430 -435 u=-3
546    0        15 -30 23 -26 410 -435 u=-3
547    0        31 -20 23 -26 410 -435 u=-3
548    0        30 -31 23 -32 410 -435 u=-3
549    0        30 -31 33 -26 410 -435 u=-3
550    0        15 -30 23 -28 435 -460 u=-3
551    0        31 -20 23 -28 435 -460 u=-3
```

```
552    0        30 -31 33 -28 435 -460 u=-3
553    0        30 -31 23 -32 435 -460 u=-3
c      fuel element
554    5 -1.51304  30 -31 32 -33     -420 u=3
555    2 -3.979996 30 -31 32 -33 420 -425 u=-3
556    0           30 -31 32 -33 425 -430 u=-3
557    5 -0.270112 30 -31 32 -33 430 -440 u=-3
558    5 -0.689740 30 -31 32 -33 440 -445 u=-3
559    5 -1.393935 30 -31 32 -33 445 -450 u=-3
560    5 -0.261853 30 -31 32 -33 450 -670 u=-3
561    5 -0.261853 30 -31 32 -33 670 -455 u=-3
562    0           30 -31 32 -33 455  u=3
563    0           -30           460 u=3
564    0            31           460 u=3
565    0        30 -31 -32       460 u=3
566    0        30 -31  33       460 u=3
c
c     universe 4
c
c      egg crate
601    0           -30           -400 u=4
602    0            31           -400 u=4
603    0        30 -31 -32       -400 u=4
604    0        30 -31  33       -400 u=4
605    5 -7.92        28     400 -420 u=4
606    5 -7.92           -23 400 -420 u=4
607    5 -7.92     -15 23 -28 400 -420 u=4
608    5 -7.92  20     23 -28 400 -420 u=4
609    5 -7.92        28     420 -430 u=4
610    5 -7.92           -23 420 -430 u=4
611    5 -7.92     -15 23 -28 420 -430 u=4
612    5 -7.92  20     23 -28 420 -430 u=4
613    5 -7.92        28     430 -440 u=4
614    5 -7.92           -23 430 -440 u=4
615    5 -7.92     -15 23 -28 430 -440 u=4
616    5 -7.92  20     23 -28 430 -440 u=4
617    5 -7.92        28     440 -670 u=4
618    5 -7.92           -23 440 -670 u=4
619    5 -7.92     -15 23 -28 440 -670 u=4
620    5 -7.92  20     23 -28 440 -670 u=4
621    5 -7.92        28     670 -460 u=4
622    5 -7.92           -23 670 -460 u=4
623    5 -7.92     -15 23 -28 670 -460 u=4
624    5 -7.92  20     23 -28 670 -460 u=4
c      boral and inside of egg crate and outside of fuel
626    0        15 -30 23 -28 400 -410 u=-4
627    0        31 -20 23 -28 400 -410 u=-4
628    0        30 -31 33 -28 400 -410 u=-4
629    0        30 -31 23 -32 400 -410 u=-4
632    0        15 -17 23 -24 410 -435 u=-4
633    0        15 -17 25 -28 410 -435 u=-4
636    6 -2.644 15 -16 24 -25 410 -420 u=-4
637    5 -7.92  16 -17 24 -25 410 -420 u=-4
640    6 -2.644 15 -16 24 -25 420 -430 u=-4
641    5 -7.92  16 -17 24 -25 420 -430 u=-4
644    6 -2.644 15 -16 24 -25 430 -435 u=-4
645    5 -7.92  16 -17 24 -25 430 -435 u=-4
646    0        17 -30 23 -28 410 -435 u=-4
647    0        31 -20 23 -28 410 -435 u=-4
648    0        30 -31 23 -32 410 -435 u=-4
649    0        30 -31 33 -28 410 -435 u=-4
```

```
650    0        15 -30 23 -28 435 -460 u=-4
651    0        31 -20 23 -28 435 -460 u=-4
652    0        30 -31 33 -28 435 -460 u=-4
653    0        30 -31 23 -32 435 -460 u=-4
c      fuel element
654    5 -1.51304  30 -31 32 -33     -420 u=4
655    2 -3.979996 30 -31 32 -33 420 -425 u=-4
656    0           30 -31 32 -33 425 -430 u=-4
657    5 -0.270112 30 -31 32 -33 430 -440 u=-4
658    5 -0.689740 30 -31 32 -33 440 -445 u=-4
659    5 -1.393935 30 -31 32 -33 445 -450 u=-4
660    5 -0.261853 30 -31 32 -33 450 -670 u=-4
661    5 -0.261853 30 -31 32 -33 670 -455 u=-4
662    0           30 -31 32 -33 455  u=4
663    0          -30            460 u=4
664    0           31            460 u=4
665    0        30 -31 -32        460 u=4
666    0        30 -31  33        460 u=4
c
c     universe 5
c
701    0              -400 u=5
702    5 -7.92 20 400 -420 u=5
703    5 -7.92 20 420 -430 u=5
704    5 -7.92 20 430 -440 u=5
705    5 -7.92 20 440 -670 u=5
706    5 -7.92 20 670 -460 u=5
707    0      -20 400 -460 u=5
708    0          460      u=5
c
c     universe 6
c
710    0                -400 u=6
711    5 -7.92  -23 400 -420 u=6
712    5 -7.92  -23 420 -430 u=6
713    5 -7.92  -23 430 -440 u=6
714    5 -7.92  -23 440 -670 u=6
715    5 -7.92  -23 670 -460 u=6
716    0         23 400 -460 u=6
717    0            460      u=6
c
c     universe 7
c
720    0                      -400 u=7
721    5 -7.92 20      23 400 -420 u=7
722    5 -7.92    -23     400 -420 u=7
723    5 -7.92 20      23 420 -430 u=7
724    5 -7.92    -23     420 -430 u=7
725    5 -7.92 20      23 430 -440 u=7
726    5 -7.92    -23     430 -440 u=7
727    5 -7.92 20      23 440 -670 u=7
728    5 -7.92    -23     440 -670 u=7
729    5 -7.92 20      23 670 -460 u=7
730    5 -7.92    -23     670 -460 u=7
731    0          -20  23 400 -460 u=7
732    0                  460      u=7
c
c     universe 8
c
735   0                          -400 u=8
736   0        15            400 -460 u=8
```

```
747   5 -7.92     -15        400 -420 u=8
748   5 -7.92     -15        420 -430 u=8
749   5 -7.92     -15        430 -440 u=8
750   5 -7.92     -15        440 -670 u=8
751   5 -7.92     -15        670 -460 u=8
752   0                      460      u=8
c
c     universe 9
c
755   0        15     23     400 -460 u=9
766   0                          -400 u=9
767   5 -7.92     -15 23     400 -420 u=9
768   5 -7.92     -23        400 -420 u=9
769   5 -7.92     -15 23     420 -430 u=9
770   5 -7.92     -23        420 -430 u=9
771   5 -7.92     -15 23     430 -440 u=9
772   5 -7.92     -23        430 -440 u=9
773   5 -7.92     -15 23     440 -670 u=9
774   5 -7.92     -23        440 -670 u=9
775   5 -7.92     -15 23     670 -460 u=9
776   5 -7.92     -23        670 -460 u=9
777   0                      460      u=9
c
c     universe 10
c
780   0        15        -28 400 -460 u=10
799   0                          -400 u=10
800   5 -7.92     -15    -28 400 -420 u=10
801   5 -7.92         28     400 -420 u=10
802   5 -7.92     -15    -28 420 -430 u=10
803   5 -7.92         28     420 -430 u=10
804   5 -7.92     -15    -28 430 -440 u=10
805   5 -7.92         28     430 -440 u=10
806   5 -7.92     -15    -28 440 -670 u=10
807   5 -7.92         28     440 -670 u=10
808   5 -7.92     -15    -28 670 -460 u=10
809   5 -7.92         28     670 -460 u=10
810   0                      460      u=10
c
c     universe 11
c
811   0                  -28 400 -460 u=11
822   0                          -400 u=11
823   5 -7.92         28     400 -420 u=11
824   5 -7.92         28     420 -430 u=11
825   5 -7.92         28     430 -440 u=11
826   5 -7.92         28     440 -670 u=11
827   5 -7.92         28     670 -460 u=11
828   0                      460      u=11
c
c     universe 12
c
830   0           -20    -28 400 -460 u=12
841   0                          -400 u=12
842   5 -7.92     -20 28     400 -420 u=12
843   5 -7.92  20            400 -420 u=12
844   5 -7.92     -20 28     420 -430 u=12
845   5 -7.92  20            420 -430 u=12
846   5 -7.92     -20 28     430 -440 u=12
847   5 -7.92  20            430 -440 u=12
848   5 -7.92     -20 28     440 -670 u=12
```

```
849   5 -7.92  20            440 -670 u=12
850   5 -7.92      -20 28     670 -460 u=12
851   5 -7.92  20            670 -460 u=12
852   0                      460      u=12
c
c     universe 13
c
c     810   0         -420  u=13
c     811   0    420  -430  u=13
c     812   0    430  -440  u=13
c     813   0    440  -670  u=13
c     814   0    670        u=13
c     storage locations
c
201   0  -301  -106 212 620 -675
202   0  -301  106 -107 212 620 -675
         fill=6 (-8.2423 90.6653 0.0)
203   0  -301  107 -108 212 620 -675
         fill=6 ( 8.2423 90.6653 0.0)
204   0  -301  108 212 620 -675
c
205   0  -301 -104 211 620 -675
206   0  -301 104 -105 211 620 -675
         fill=6 (-41.2115 74.1807 0.0)
207   0  -301 105 -106 211 -212 620 -675
         fill=7 (-24.7269 74.1807 0.0)
c
101   0  106 -107 211 -212  620 -675
         fill=2 (-8.2423 74.1807 0.0)
102   0  107 -108 211 -212  620 -675
         fill=4 ( 8.2423 74.1807 0.0)
c
208   0  -301 108 -109 211 -212 620 -675
         fill=9 (24.7269 74.1807 0.0)
209   0  -301 109 -110 211 620 -675
         fill=6 (41.2115 74.1807 0.0)
210   0  -301 110 211 620 -675
c
211   0  -301 -103 210 620 -675
212   0  -301 103 -104 210 -211 620 -675
         fill=7 (-57.6961 57.6961 0.0)
c
103   0  104 -105 210 -211  620 -675
         fill=2 (-41.2115 57.6961 0.0)
104   0  105 -106 210 -211  620 -675
         fill=4 (-24.7269 57.6961 0.0)
105   0  106 -107 210 -211  620 -675
         fill=1 (-8.2423 57.6961 0.0)
106   0  107 -108 210 -211  620 -675
         fill=1 ( 8.2423 57.6961 0.0)
107   0  108 -109 210 -211  620 -675
         fill=4 ( 24.7269 57.6961 0.0)
108   0  109 -110 210 -211  620 -675
         fill=4 ( 41.2115 57.6961 0.0)
c
213   0  -301 110 -111 210 -211 620 -675
         fill=9 ( 57.6961 57.6961 0.0)
214   0  -301 111 210 620 -675
c
215   0  -301 -103 209 -210 620 -675
         fill=5 (-74.1807 41.2115 0.0)
```

```
c
109   0  103 -104 209 -210  620 -675
         fill=2 (-57.6961 41.2115 0.0)
110   0  104 -105 209 -210  620 -675
         fill=1 (-41.2115 41.2115 0.0)
111   0  105 -106 209 -210  620 -675
         fill=1 (-24.7269 41.2115 0.0)
112   0  106 -107 209 -210  620 -675
         fill=1 (-8.2423 41.2115 0.0)
113   0  107 -108 209 -210  620 -675
         fill=1 ( 8.2423 41.2115 0.0)
114   0  108 -109 209 -210  620 -675
         fill=1 ( 24.7269 41.2115 0.0)
115   0  109 -110 209 -210  620 -675
         fill=1 ( 41.2115 41.2115 0.0)
116   0  110 -111 209 -210  620 -675
         fill=4 ( 57.6961 41.2115 0.0)
c
216   0  -301 111 209 -210 620 -675
         fill=8 (74.1807 41.2115 0.0)
c
217   0  -301 -102 208 620 -675
218   0  -301 102 -103 208 -209 620 -675
         fill=7 (-74.1807 24.7269 0.0)
c
117   0  103 -104 208 -209  620 -675
         fill=3 (-57.6961 24.7269 0.0)
118   0  104 -105 208 -209  620 -675
         fill=1 (-41.2115 24.7269 0.0)
119   0  105 -106 208 -209  620 -675
         fill=1 (-24.7269 24.7269 0.0)
120   0  106 -107 208 -209  620 -675
         fill=1 (-8.2423 24.7269 0.0)
121   0  107 -108 208 -209  620 -675
         fill=1 ( 8.2423 24.7269 0.0)
122   0  108 -109 208 -209  620 -675
         fill=1 ( 24.7269 24.7269 0.0)
123   0  109 -110 208 -209  620 -675
         fill=1 ( 41.2115 24.7269 0.0)
124   0  110 -111 208 -209  620 -675
         fill=1 ( 57.6961 24.7269 0.0)
c
219   0  -301 111 -112 208 -209 620 -675
         fill=9 (74.1807 24.7269 0.0)
220   0  -301 112 208 620 -675
c
221   0  -301 -102 207 -208 620 -675
         fill=5 (-90.6653 8.2423 0.0)
c
125   0  102 -103 207 -208  620 -675
         fill=2 (-74.1807 8.2423 0.0)
126   0  103 -104 207 -208  620 -675
         fill=1 (-57.6961 8.2423 0.0)
127   0  104 -105 207 -208  620 -675
         fill=1 (-41.2115 8.2423 0.0)
128   0  105 -106 207 -208  620 -675
         fill=1 (-24.7269 8.2423 0.0)
129   0  106 -107 207 -208  620 -675
         fill=1 (-8.2423 8.2423 0.0)
130   0  107 -108 207 -208  620 -675
         fill=1 ( 8.2423 8.2423 0.0)
```

```
131   0  108 -109 207 -208  620 -675
         fill=1 ( 24.7269 8.2423 0.0)
132   0  109 -110 207 -208  620 -675
         fill=1 ( 41.2115 8.2423 0.0)
133   0  110 -111 207 -208  620 -675
         fill=1 ( 57.6961 8.2423 0.0)
134   0  111 -112 207 -208  620 -675
         fill=4 ( 74.1807 8.2423 0.0)
c
222   0  -301 112 207 -208 620 -675
         fill=8 (90.6653 8.2423 0.0)
c
223   0  -301 -102 206 -207 620 -675
         fill=5 (-90.6653 -8.2423 0.0)
c
135   0  102 -103 206 -207  620 -675
         fill=3 (-74.1807 -8.2423 0.0)
136   0  103 -104 206 -207  620 -675
         fill=1 (-57.6961 -8.2423 0.0)
137   0  104 -105 206 -207  620 -675
         fill=1 (-41.2115 -8.2423 0.0)
138   0  105 -106 206 -207  620 -675
         fill=1 (-24.7269 -8.2423 0.0)
139   0  106 -107 206 -207  620 -675
         fill=1 (-8.2423 -8.2423 0.0)
140   0  107 -108 206 -207  620 -675
         fill=1 ( 8.2423 -8.2423 0.0)
141   0  108 -109 206 -207  620 -675
         fill=1 ( 24.7269 -8.2423 0.0)
142   0  109 -110 206 -207  620 -675
         fill=1 ( 41.2115 -8.2423 0.0)
143   0  110 -111 206 -207  620 -675
         fill=1 ( 57.6961 -8.2423 0.0)
144   0  111 -112 206 -207  620 -675
         fill=1 ( 74.1807 -8.2423 0.0)
c
224   0  -301 112 206 -207 620 -675
         fill=8 (90.6653 -8.2423 0.0)
c
225   0  -301 -102 -206 620 -675
226   0  -301 102 -103 205 -206 620 -675
         fill=12 (-74.1807 -24.7269 0.0)
c
145   0  103 -104 205 -206  620 -675
         fill=3 (-57.6961 -24.7269 0.0)
146   0  104 -105 205 -206  620 -675
         fill=1 (-41.2115 -24.7269 0.0)
147   0  105 -106 205 -206  620 -675
         fill=1 (-24.7269 -24.7269 0.0)
148   0  106 -107 205 -206  620 -675
         fill=1 (-8.2423  -24.7269 0.0)
149   0  107 -108 205 -206  620 -675
         fill=1 ( 8.2423  -24.7269 0.0)
150   0  108 -109 205 -206  620 -675
         fill=1 ( 24.7269 -24.7269 0.0)
151   0  109 -110 205 -206  620 -675
         fill=1 ( 41.2115 -24.7269 0.0)
152   0  110 -111 205 -206  620 -675
         fill=1 ( 57.6961 -24.7269 0.0)
c
227   0  -301 111 -112 205 -206 620 -675
```

```
          fill=10 (74.1807 -24.7269 0.0)
228   0  -301 112 -206 620 -675
c
229   0  -301 -103 204 -205 620 -675
          fill=5 (-74.1807 -41.2115 0.0)
c
153   0  103 -104 204 -205  620 -675
          fill=3 (-57.6961 -41.2115 0.0)
154   0  104 -105 204 -205  620 -675
          fill=1 (-41.2115 -41.2115 0.0)
155   0  105 -106 204 -205  620 -675
          fill=1 (-24.7269 -41.2115 0.0)
156   0  106 -107 204 -205  620 -675
          fill=1 (-8.2423  -41.2115 0.0)
157   0  107 -108 204 -205  620 -675
          fill=1 ( 8.2423  -41.2115 0.0)
158   0  108 -109 204 -205  620 -675
          fill=1 ( 24.7269 -41.2115 0.0)
159   0  109 -110 204 -205  620 -675
          fill=1 ( 41.2115 -41.2115 0.0)
160   0  110 -111 204 -205  620 -675
          fill=1 ( 57.6961 -41.2115 0.0)
c
230   0  -301 111 204 -205 620 -675
          fill=8 (74.1807 -41.2115 0.0)
c
231   0  -301 -103 -204 620 -675
232   0  -301 103 -104 203 -204 620 -675
          fill=12 (-57.6961 -57.6961 0.0)
c
161   0  104 -105 203 -204  620 -675
          fill=3 (-41.2115 -57.6961 0.0)
162   0  105 -106 203 -204  620 -675
          fill=1 (-24.7269 -57.6961 0.0)
163   0  106 -107 203 -204  620 -675
          fill=1 (-8.2423 -57.6961 0.0)
164   0  107 -108 203 -204  620 -675
          fill=1 ( 8.2423 -57.6961 0.0)
165   0  108 -109 203 -204  620 -675
          fill=1 ( 24.7269 -57.6961 0.0)
166   0  109 -110 203 -204  620 -675
          fill=1 ( 41.2115 -57.6961 0.0)
c
233   0  -301 110 -111 203 -204 620 -675
          fill=10 ( 57.6961 -57.6961 0.0)
234   0  -301 111 -204 620 -675
c
235   0  -301 -104 -203 620 -675
236   0  -301 104 -105 -203 620 -675
          fill=11 (-41.2115 -74.1807 0.0)
237   0  -301 105 -106 202 -203 620 -675
          fill=12 (-24.7269 -74.1807 0.0)
c
167   0  106 -107 202 -203  620 -675
          fill=3 (-8.2423 -74.1807 0.0)
168   0  107 -108 202 -203  620 -675
          fill=1 ( 8.2423 -74.1807 0.0)
c
238   0  -301 108 -109 202 -203 620 -675
          fill=10 (24.7269 -74.1807 0.0)
239   0  -301 109 -110 -203 620 -675
```

```
         fill=11 (41.2115 -74.1807 0.0)
240   0  -301 110 -203 620 -675
c
241   0  -301  -106 -202 620 -675
242   0  -301  106 -107 -202 620 -675
         fill=11 (-8.2423 -90.6653 0.0)
243   0  -301  107 -108 -202 620 -675
         fill=11 ( 8.2423 -90.6653 0.0)
244   0  -301  108 -202 620 -675
c
1821   5 -7.92 301 -302 610 -615  $ MPC shell
1822   0       302 -501 610 -615  $ Air gap
1823   5 -7.92 301 -302 615 -630  $ MPC shell
1824   0       302 -501 615 -630  $ Air gap
1825   5 -7.92 301 -302 630 -420  $ MPC shell
1826   0       302 -501 630 -420  $ Air gap
1827   5 -7.92 301 -302 420 -430  $ MPC shell
1828   0       302 -501 420 -430  $ Air gap
1829   5 -7.92 301 -302 430 -440  $ MPC shell
1830   0       302 -501 430 -440  $ Air gap
1831   5 -7.92 301 -302 440 -670  $ MPC shell
1832   0       302 -501 440 -670  $ Air gap
1833   5 -7.92 301 -302 670 -675  $ MPC shell
1834   0       302 -501 670 -675  $ Air gap
1835   5 -7.92 301 -302 675 -651  $ MPC shell
1836   0       302 -501 675 -651  $ Air gap
1837   5 -7.92 301 -302 651 -652  $ MPC shell
1838   0       302 -501 651 -652  $ Air gap
1839   5 -7.92 301 -302 652 -653  $ MPC shell
1840   0       302 -501 652 -653  $ Air gap
1841   5 -7.92 301 -302 653 -654  $ MPC shell
1842   0       302 -501 653 -654  $ Air gap
1843   5 -7.92 301 -302 654 -655  $ MPC shell
1844   0       302 -501 654 -655  $ Air gap
1845   5 -7.92 301 -302 655 -656  $ MPC shell
1846   0       302 -501 655 -656  $ Air gap
1847   5 -7.92 301 -302 656 -657  $ MPC shell
1848   0       302 -501 656 -657  $ Air gap
1849   5 -7.92 301 -302 657 -680  $ MPC shell
1850   0       301 -302 680 -685  $ Air gap
1851   0       302 -501 657 -685  $ Air gap
c
1854   5 -7.92     -301 610 -615  $ MPC baseplate
1855   5 -7.92     -301 615 -620  $ MPC baseplate
1860   5 -7.92     -301 675 -651  $ MPC lid (both)
1861   5 -7.92     -301 651 -652  $ MPC lid (both)
1862   5 -7.92     -301 652 -653  $ MPC lid (both)
1863   5 -7.92     -301 653 -654  $ MPC lid (both)
1864   5 -7.92     -301 654 -655  $ MPC lid (both)
1865   5 -7.92     -301 655 -656  $ MPC lid (both)
1866   5 -7.92     -301 656 -657  $ MPC lid (both)
1867   5 -7.92     -301 657 -680  $ MPC lid (both)
1868   0           -301 680 -685  $ Air gap
c     OVERPACK \/ \/ \/ \/ \/
c
1001  8 -7.82 501 -508 630 -420 $ steel shell
1002  8 -7.82 501 -508 420 -430 $ steel shell
1003  8 -7.82 501 -508 430 -440 $ steel shell
1004  8 -7.82 501 -508 440 -670 $ steel shell
1005  0       508 -512 630 -420  fill=20
1006  0       508 -512 420 -430  fill=20
```

```
1007  0       508 -512 430 -440  fill=20
1008  0       508 -512 440 -670  fill=20
1009  8 -7.82 512 -513 630 -420 $ outer steel shell
1010  8 -7.82 512 -513 420 -430 $ outer steel shell
1011  8 -7.82 512 -513 430 -440 $ outer steel shell
1012  8 -7.82 512 -513 440 -670 $ outer steel shell
c     1012  8 -7.82  512 -513 640 -670             $ outer steel shell
c     1013  0        512 -513 630 -640  1103 -1102 $ air in pocket trun.
c     1014  0        512 -513 630 -640  2103 -2102 $ air in pocket trun.
c     1015  8 -7.82  512 -513 630 -640  1102  2102 $ outer steel shell
c     1016  8 -7.82  512 -513 630 -640  1102 -2103 $ outer steel shell
c     1017  8 -7.82  512 -513 630 -640 -1103 -2103 $ outer steel shell
c     1018  8 -7.82  512 -513 630 -640 -1103  2102 $ outer steel shell
c
c      steel spines and holtite
10101   8 -7.82   2000 -2002  645 -660   1000  u=20 $ steel spine
10102   7 -1.61   2002 -2011  645 -660   2000  u=20 $ holtite
10103   8 -7.82   2011 -2012  645 -660   2000  u=20 $ steel spine
10104   7 -1.61   2012 -2021  645 -660   2000  u=20 $ holtite
10105   8 -7.82   2021 -2022  645 -660   2000  u=20 $ steel spine
10106   7 -1.61   2022 -2031  645 -660   2000  u=20 $ holtite
10107   8 -7.82   2031 -2032  645 -660   2000  u=20 $ steel spine
10108   7 -1.61   2032 -2041  645 -660   2000  u=20 $ holtite
10109   8 -7.82   2041 -2042  645 -660   2000  u=20 $ steel spine
10110   7 -1.61   2042 -2051  645 -660   2000  u=20 $ holtite
10111   8 -7.82   2051 -2052  645 -660   2000  u=20 $ steel spine
10112   7 -1.61   2052 -2061  645 -660   2000  u=20 $ holtite
10113   8 -7.82   2061 -2062  645 -660   2000  u=20 $ steel spine
10114   7 -1.61   2062 -2071  645 -660   2000  u=20 $ holtite
10115   8 -7.82   2071 -2072  645 -660   2000  u=20 $ steel spine
10116   7 -1.61   2072 -2081  645 -660   2000  u=20 $ holtite
10117   8 -7.82   2081 -2082  645 -660   2000  u=20 $ steel spine
10118   7 -1.61   2082 -2091  645 -660   2000  u=20 $ holtite
10119   8 -7.82   2091 -2092  645 -660   2000  u=20 $ steel spine
10120   7 -1.61   2092  1002  645 -660   2000  u=20 $ holtite
10121   8 -7.82   1000 -1002  645 -660   2000  u=20 $ steel spine
c
10122   8 -7.82   1001 -1000  645 -660   2000  u=20 $ steel spine
10123   7 -1.61   1012 -1001  645 -660   2000  u=20 $ holtite
10124   8 -7.82   1011 -1012  645 -660   2000  u=20 $ steel spine
10125   7 -1.61   1022 -1011  645 -660   2000  u=20 $ holtite
10126   8 -7.82   1021 -1022  645 -660   2000  u=20 $ steel spine
10127   7 -1.61   1032 -1021  645 -660   2000  u=20 $ holtite
10128   8 -7.82   1031 -1032  645 -660   2000  u=20 $ steel spine
10129   7 -1.61   1042 -1031  645 -660   2000  u=20 $ holtite
10130   8 -7.82   1041 -1042  645 -660   2000  u=20 $ steel spine
10131   7 -1.61   1052 -1041  645 -660   2000  u=20 $ holtite
10132   8 -7.82   1051 -1052  645 -660   2000  u=20 $ steel spine
10133   7 -1.61   1062 -1051  645 -660   2000  u=20 $ holtite
10134   8 -7.82   1061 -1062  645 -660   2000  u=20 $ steel spine
10135   7 -1.61   1072 -1061  645 -660   2000  u=20 $ holtite
10136   8 -7.82   1071 -1072  645 -660   2000  u=20 $ steel spine
10137   7 -1.61   1082 -1071  645 -660   2000  u=20 $ holtite
10138   8 -7.82   1081 -1082  645 -660   2000  u=20 $ steel spine
10139   7 -1.61   1092 -1081  645 -660   2000  u=20 $ holtite
10140   8 -7.82   1091 -1092  645 -660   2000  u=20 $ steel spine
10141   7 -1.61   2002 -1091  645 -660   2000  u=20 $ holtite
10142   8 -7.82   2000 -2002  645 -660  -1000  u=20 $ steel spine
c
10143   8 -7.82   2001 -2000  645 -660  -1000  u=20 $ steel spine
10144   7 -1.61   2012 -2001  645 -660  -2000  u=20 $ holtite
```

```
10145   8 -7.82   2011 -2012  645 -660  -2000  u=20 $ steel spine
10146   7 -1.61   2022 -2011  645 -660  -2000  u=20 $ holtite
10147   8 -7.82   2021 -2022  645 -660  -2000  u=20 $ steel spine
10148   7 -1.61   2032 -2021  645 -660  -2000  u=20 $ holtite
10149   8 -7.82   2031 -2032  645 -660  -2000  u=20 $ steel spine
10150   7 -1.61   2042 -2031  645 -660  -2000  u=20 $ holtite
10151   8 -7.82   2041 -2042  645 -660  -2000  u=20 $ steel spine
10152   7 -1.61   2052 -2041  645 -660  -2000  u=20 $ holtite
10153   8 -7.82   2051 -2052  645 -660  -2000  u=20 $ steel spine
10154   7 -1.61   2062 -2051  645 -660  -2000  u=20 $ holtite
10155   8 -7.82   2061 -2062  645 -660  -2000  u=20 $ steel spine
10156   7 -1.61   2072 -2061  645 -660  -2000  u=20 $ holtite
10157   8 -7.82   2071 -2072  645 -660  -2000  u=20 $ steel spine
10158   7 -1.61   2082 -2071  645 -660  -2000  u=20 $ holtite
10159   8 -7.82   2081 -2082  645 -660  -2000  u=20 $ steel spine
10160   7 -1.61   2092 -2081  645 -660  -2000  u=20 $ holtite
10161   8 -7.82   2091 -2092  645 -660  -2000  u=20 $ steel spine
10162   7 -1.61  -1001 -2091  645 -660  -2000  u=20 $ holtite
10163   8 -7.82   1001 -1000  645 -660  -2000  u=20 $ steel spine
c
10164   8 -7.82   1000 -1002  645 -660  -2000  u=20 $ steel spine
10165   7 -1.61   1002 -1011  645 -660  -2000  u=20 $ holtite
10166   8 -7.82   1011 -1012  645 -660  -2000  u=20 $ steel spine
10167   7 -1.61   1012 -1021  645 -660  -2000  u=20 $ holtite
10168   8 -7.82   1021 -1022  645 -660  -2000  u=20 $ steel spine
10169   7 -1.61   1022 -1031  645 -660  -2000  u=20 $ holtite
10170   8 -7.82   1031 -1032  645 -660  -2000  u=20 $ steel spine
10171   7 -1.61   1032 -1041  645 -660  -2000  u=20 $ holtite
10172   8 -7.82   1041 -1042  645 -660  -2000  u=20 $ steel spine
10173   7 -1.61   1042 -1051  645 -660  -2000  u=20 $ holtite
10174   8 -7.82   1051 -1052  645 -660  -2000  u=20 $ steel spine
10175   7 -1.61   1052 -1061  645 -660  -2000  u=20 $ holtite
10176   8 -7.82   1061 -1062  645 -660  -2000  u=20 $ steel spine
10177   7 -1.61   1062 -1071  645 -660  -2000  u=20 $ holtite
10178   8 -7.82   1071 -1072  645 -660  -2000  u=20 $ steel spine
10179   7 -1.61   1072 -1081  645 -660  -2000  u=20 $ holtite
10180   8 -7.82   1081 -1082  645 -660  -2000  u=20 $ steel spine
10181   7 -1.61   1082 -1091  645 -660  -2000  u=20 $ holtite
10182   8 -7.82   1091 -1092  645 -660  -2000  u=20 $ steel spine
10183   7 -1.61   1092 -2001  645 -660  -2000  u=20 $ holtite
10184   8 -7.82   2001 -2000  645 -660   1000  u=20 $ steel spine
c
c    10201   8 -7.82   2000 -2002  635 -645   1000  u=20 $ steel spine
c    10202   7 -1.61   2002 -2011  635 -645   2000  u=20 $ holtite
10202   7 -1.61   2101 -2011  635 -645   2000  u=20 $ holtite
10203   8 -7.82   2011 -2012  635 -645   2000  u=20 $ steel spine
10204   7 -1.61   2012 -2021  635 -645   2000  u=20 $ holtite
10205   8 -7.82   2021 -2022  635 -645   2000  u=20 $ steel spine
10206   7 -1.61   2022 -2031  635 -645   2000  u=20 $ holtite
10207   8 -7.82   2031 -2032  635 -645   2000  u=20 $ steel spine
10208   7 -1.61   2032 -2041  635 -645   2000  u=20 $ holtite
10209   8 -7.82   2041 -2042  635 -645   2000  u=20 $ steel spine
10210   7 -1.61   2042 -2051  635 -645   2000  u=20 $ holtite
10211   8 -7.82   2051 -2052  635 -645   2000  u=20 $ steel spine
10212   7 -1.61   2052 -2061  635 -645   2000  u=20 $ holtite
10213   8 -7.82   2061 -2062  635 -645   2000  u=20 $ steel spine
10214   7 -1.61   2062 -2071  635 -645   2000  u=20 $ holtite
10215   8 -7.82   2071 -2072  635 -645   2000  u=20 $ steel spine
10216   7 -1.61   2072 -2081  635 -645   2000  u=20 $ holtite
10217   8 -7.82   2081 -2082  635 -645   2000  u=20 $ steel spine
10218   7 -1.61   2082 -2091  635 -645   2000  u=20 $ holtite
```

```
10219   8 -7.82   2091 -2092  635 -645   2000  u=20 $ steel spine
c    10220   7 -1.61   2092  1101  635 -645   2000  u=20 $ holtite
10220   7 -1.61   2092  1002  635 -645   2000  u=20 $ holtite
10221   8 -7.82   1000 -1002  635 -645   2000  u=20 $ steel spine
c
10222   8 -7.82   1001 -1000  635 -645   2000  u=20 $ steel spine
10223   7 -1.61   1012 -1001  635 -645   2000  u=20 $ holtite
c    10223   7 -1.61   1012 -1104  635 -645   2000  u=20 $ holtite
10224   8 -7.82   1011 -1012  635 -645   2000  u=20 $ steel spine
10225   7 -1.61   1022 -1011  635 -645   2000  u=20 $ holtite
10226   8 -7.82   1021 -1022  635 -645   2000  u=20 $ steel spine
10227   7 -1.61   1032 -1021  635 -645   2000  u=20 $ holtite
10228   8 -7.82   1031 -1032  635 -645   2000  u=20 $ steel spine
10229   7 -1.61   1042 -1031  635 -645   2000  u=20 $ holtite
10230   8 -7.82   1041 -1042  635 -645   2000  u=20 $ steel spine
10231   7 -1.61   1052 -1041  635 -645   2000  u=20 $ holtite
10232   8 -7.82   1051 -1052  635 -645   2000  u=20 $ steel spine
10233   7 -1.61   1062 -1051  635 -645   2000  u=20 $ holtite
10234   8 -7.82   1061 -1062  635 -645   2000  u=20 $ steel spine
10235   7 -1.61   1072 -1061  635 -645   2000  u=20 $ holtite
10236   8 -7.82   1071 -1072  635 -645   2000  u=20 $ steel spine
10237   7 -1.61   1082 -1071  635 -645   2000  u=20 $ holtite
10238   8 -7.82   1081 -1082  635 -645   2000  u=20 $ steel spine
10239   7 -1.61   1092 -1081  635 -645   2000  u=20 $ holtite
10240   8 -7.82   1091 -1092  635 -645   2000  u=20 $ steel spine
10241   7 -1.61   2101 -1091  635 -645   2000  u=20 $ holtite
c    10241   7 -1.61   2002 -1091  635 -645   2000  u=20 $ holtite
c    10242   8 -7.82   2000 -2002  635 -645  -1000  u=20 $ steel spine
c
c    10243   8 -7.82   2001 -2000  635 -645  -1000  u=20 $ steel spine
c    10244   7 -1.61   2012 -2001  635 -645  -2000  u=20 $ holtite
10244   7 -1.61   2012 -2104  635 -645  -2000  u=20 $ holtite
10245   8 -7.82   2011 -2012  635 -645  -2000  u=20 $ steel spine
10246   7 -1.61   2022 -2011  635 -645  -2000  u=20 $ holtite
10247   8 -7.82   2021 -2022  635 -645  -2000  u=20 $ steel spine
10248   7 -1.61   2032 -2021  635 -645  -2000  u=20 $ holtite
10249   8 -7.82   2031 -2032  635 -645  -2000  u=20 $ steel spine
10250   7 -1.61   2042 -2031  635 -645  -2000  u=20 $ holtite
10251   8 -7.82   2041 -2042  635 -645  -2000  u=20 $ steel spine
10252   7 -1.61   2052 -2041  635 -645  -2000  u=20 $ holtite
10253   8 -7.82   2051 -2052  635 -645  -2000  u=20 $ steel spine
10254   7 -1.61   2062 -2051  635 -645  -2000  u=20 $ holtite
10255   8 -7.82   2061 -2062  635 -645  -2000  u=20 $ steel spine
10256   7 -1.61   2072 -2061  635 -645  -2000  u=20 $ holtite
10257   8 -7.82   2071 -2072  635 -645  -2000  u=20 $ steel spine
10258   7 -1.61   2082 -2071  635 -645  -2000  u=20 $ holtite
10259   8 -7.82   2081 -2082  635 -645  -2000  u=20 $ steel spine
10260   7 -1.61   2092 -2081  635 -645  -2000  u=20 $ holtite
10261   8 -7.82   2091 -2092  635 -645  -2000  u=20 $ steel spine
c    10262   7 -1.61  -1104 -2091  635 -645  -2000  u=20 $ holtite
10262   7 -1.61  -1001 -2091  635 -645  -2000  u=20 $ holtite
10263   8 -7.82   1001 -1000  635 -645  -2000  u=20 $ steel spine
c
10264   8 -7.82   1000 -1002  635 -645  -2000  u=20 $ steel spine
10265   7 -1.61   1002 -1011  635 -645  -2000  u=20 $ holtite
c    10265   7 -1.61   1101 -1011  635 -645  -2000  u=20 $ holtite
10266   8 -7.82   1011 -1012  635 -645  -2000  u=20 $ steel spine
10267   7 -1.61   1012 -1021  635 -645  -2000  u=20 $ holtite
10268   8 -7.82   1021 -1022  635 -645  -2000  u=20 $ steel spine
10269   7 -1.61   1022 -1031  635 -645  -2000  u=20 $ holtite
10270   8 -7.82   1031 -1032  635 -645  -2000  u=20 $ steel spine
```

```
10271   7 -1.61   1032 -1041  635 -645  -2000  u=20 $ holtite
10272   8 -7.82   1041 -1042  635 -645  -2000  u=20 $ steel spine
10273   7 -1.61   1042 -1051  635 -645  -2000  u=20 $ holtite
10274   8 -7.82   1051 -1052  635 -645  -2000  u=20 $ steel spine
10275   7 -1.61   1052 -1061  635 -645  -2000  u=20 $ holtite
10276   8 -7.82   1061 -1062  635 -645  -2000  u=20 $ steel spine
10277   7 -1.61   1062 -1071  635 -645  -2000  u=20 $ holtite
10278   8 -7.82   1071 -1072  635 -645  -2000  u=20 $ steel spine
10279   7 -1.61   1072 -1081  635 -645  -2000  u=20 $ holtite
10280   8 -7.82   1081 -1082  635 -645  -2000  u=20 $ steel spine
10281   7 -1.61   1082 -1091  635 -645  -2000  u=20 $ holtite
10282   8 -7.82   1091 -1092  635 -645  -2000  u=20 $ steel spine
10283   7 -1.61   1092 -2104  635 -645  -2000  u=20 $ holtite
c    10283   7 -1.61    1092 -2001  635 -645  -2000  u=20 $ holtite
c    10284   8 -7.82   2001 -2000  635 -645   1000  u=20 $ steel spine
c
11000   0         660 -665   u=20 $ foam
11001   8 -7.82   665        u=20 $ item 17 top
c    11002   8 -7.82   1101  2101  -635  u=20 $ item 17 bot
c    11003   8 -7.82  -2104  1101  -635  u=20 $ item 17 bot
c    11004   8 -7.82  -1104 -2104  -635  u=20 $ item 17 bot
c    11005   8 -7.82   2101 -1104  -635  u=20 $ item 17 bot
11002   8 -7.82         2101  -635  u=20 $ item 17 bot
11003   8 -7.82  -2104        -635  u=20 $ item 17 bot
c
c    pocket trunion
c
c    11101   8 -7.82  -514  1104 -1101 -640  u=20 $ pocket trunion before hole
c    11102   8 -7.82   514  1104 -1103 -640  u=20 $ steel on side of hole
c    11103   8 -7.82   514  1103 -1102 -640  u=20 $ hole
c    11104   8 -7.82   514  1102 -1101 -640  u=20 $ steel on side of hole
c    11105   8 -7.82        1104 -1101  640 -645  u=20 $ steel above hole
c
11111   8 -7.82  -514  2104 -2101 -640  u=20 $ pocket trunion before hole
11112   8 -7.82   514  2104 -2103 -640  u=20 $ steel on side of hole
11113   8 -7.82   514  2103 -2102 -640  u=20 $ hole
11114   8 -7.82   514  2102 -2101 -640  u=20 $ steel on side of hole
11115   8 -7.82        2104 -2101  640 -645  u=20 $ steel above hole
c
c     overpack base plate
c
2000  8 -7.82     -501 600 -601
2001  8 -7.82     -501 601 -602
2002  8 -7.82     -501 602 -603
2003  8 -7.82     -501 603 -604
2004  8 -7.82     -501 604 -610
c
2010  8 -7.82 501 -515 600 -601
2011  8 -7.82 501 -515 601 -602
2012  8 -7.82 501 -515 602 -603
2013  8 -7.82 501 -515 603 -604
2014  8 -7.82 501 -515 604 -610
2015  8 -7.82 501 -515 610 -615
2016  8 -7.82 501 -515 615 -630
2020  0   515 -513 600 -601
2021  0   515 -513 601 -602
2022  0   515 -513 602 -603
2023  0   515 -513 603 -604
2024  0   515 -513 604 -610
2025  0   515 -513 610 -615
2026  0   515 -513 615 -630
```

```
c
c     overpack lid
c
3000  8 -7.82     -506 685 -686
3001  8 -7.82     -506 686 -687
3002  8 -7.82     -506 687 -688
3003  8 -7.82     -506 688 -689
3004  8 -7.82     -506 689 -695
3010  0   506 -513 685 -686
3011  0   506 -513 686 -687
3012  0   506 -513 687 -688
3013  0   506 -513 688 -689
3014  0   506 -513 689 -695
c
3020  8 -7.82 501 -517 675 -676
3021  8 -7.82 501 -516 676 -651
3022  8 -7.82 501 -516 651 -652
3023  8 -7.82 501 -516 652 -653
3024  8 -7.82 501 -516 653 -654
3025  8 -7.82 501 -516 654 -655
3026  8 -7.82 501 -516 655 -656
3027  8 -7.82 501 -516 656 -677
3028  8 -7.82 501 -506 677 -657
3029  8 -7.82 501 -506 657 -685
3030  0   517 -513 675 -676
3031  0   516 -513 676 -651
3032  0   516 -513 651 -652
3033  0   516 -513 652 -653
3034  0   516 -513 653 -654
3035  0   516 -513 654 -655
3036  0   516 -513 655 -656
3037  0   516 -513 656 -677
3038  0   506 -513 677 -657
3039  0   506 -513 657 -685
c
3042  8 -7.82 501 -517 670 -675
3047  0   517 -513 670 -675
c
c     surrounding air regions
9000  9 -1.17e-3       -506 695 -830
9001  9 -1.17e-3       -506 830 -831
9002  9 -1.17e-3       -506 831 -832
9003  9 -1.17e-3       -506 832 -833
9004  9 -1.17e-3       -506 833 -837
9010  0   506 -527 695 -830
9011  0   506 -527 830 -831
9012  0   506 -527 831 -832
9013  0   506 -527 832 -833
9014  0   506 -527 833 -837
c
9100  9 -1.17e-3       -515 810 -600
9101  9 -1.17e-3       -515 811 -810
9102  9 -1.17e-3       -515 812 -811
9103  9 -1.17e-3       -515 813 -812
9104  9 -1.17e-3       -515 817 -813
9110  0   515 -527 810 -600
9111  0   515 -527 811 -810
9112  0   515 -527 812 -811
9113  0   515 -527 813 -812
9114  0   515 -527 817 -813
c
```

```
9200  0   513 -527 600 -601
9201  0   513 -527 601 -602
9202  0   513 -527 602 -603
9203  0   513 -527 603 -604
9204  0   513 -527 604 -610
9205  0   513 -527 610 -615
9206  0   513 -527 615 -630
9207  0   513 -527 630 -420
9208  0   513 -527 420 -430
9209  0   513 -527 430 -440
9210  0   513 -527 440 -670
9211  0   513 -527 670 -675
9212  0   513 -527 675 -651
9213  0   513 -527 651 -652
9214  0   513 -527 652 -653
9215  0   513 -527 653 -654
9216  0   513 -527 654 -655
9217  0   513 -527 655 -656
9218  0   513 -527 656 -657
9219  0   513 -527 657 -685
9220  0   513 -527 685 -686
9221  0   513 -527 686 -687
9222  0   513 -527 687 -688
9223  0   513 -527 688 -689
9224  0   513 -527 689 -695
c
c
c
9999  0  527:-817:837
c
c     BLANK LINE

c     BLANK LINE
c
c     MPC surfaces\/ \/ \/ \/ \/
c
1     cz         0.52832
2     cz         0.53213
3     cz         0.61341
4     cz         0.67437
5     cz         0.75057
6     px         0.8128
7     px        -0.8128
8     py         0.8128
9     py        -0.8128
10    px        -4.445
11    px         4.445
12    py        -4.445
13    py         4.445
c     14    px        -8.2423
14    px        -8.242301
15    px        -7.9248
16    px        -7.66826
17    px        -7.47776
18    px        -6.0325
19    px         6.0325
20    px         7.9248
c     21    px         8.2423
c     22    py        -8.2423
21    px         8.242301
22    py        -8.242301
```

```
23    py        -7.9248
24    py        -6.0325
25    py         6.0325
26    py         7.47776
27    py         7.66826
28    py         7.9248
c     29    py         8.2423
29    py         8.242301
c
30    px   -6.5024
31    px    6.5024
32    py   -6.5024
33    py    6.5024
c
101   px  -98.9076
102   px  -82.423
103   px  -65.9384
104   px  -49.4538
105   px  -32.9692
106   px  -16.4846
107   px   0.0
108   px   16.4846
109   px   32.9692
110   px   49.4538
111   px   65.9384
112   px   82.423
113   px   98.9076
c
201   py  -98.9076
202   py  -82.423
203   py  -65.9384
204   py  -49.4538
205   py  -32.9692
206   py  -16.4846
207   py   0.0
208   py   16.4846
209   py   32.9692
210   py   49.4538
211   py   65.9384
212   py   82.423
213   py   98.9076
c
301   cz   85.4075
302   cz   86.6775
c
c     620   pz    21.59     $ MPC baseplate        - 2.5 inches
c     400   pz   24.765  $ start of egg crate
400   pz   23.876  $ start of egg crate
410   pz   33.9725  $ start of boral
420   pz   40.3479 $ beginning of fuel
425   pz   406.1079 $ end of fuel
430   pz   421.3479 $ space
435   pz   430.2125  $ end  of boral
440   pz   445.4271 $ plenum
445   pz   448.8561 $ expansion springs
450   pz   457.3397 $ top end fitting
455   pz   468.63   $ top of element
460   pz   466.344  $ top of egg crate
c
c     MPC surfaces/\ /\ /\ /\ /\
c
```

```
c
c     OVERPACK survaces \/ \/ \/ \/ \/
c
501   cz    87.3125  $ IR for overpack
502   cz    90.4875  $ item 2  1.25 inch
503   cz    93.6625  $ item 2  1.25 inch
504   cz    96.8375    $ item 12  1.25 inch
505   cz    100.0125   $ item 13  1.25 inch
506   cz    103.1875   $ item 14  1.25 inch
507   cz    106.3625   $ item 15  1.25 inch
508   cz    108.9025   $ item 16  1 inch
509   cz    111.521875  $ holtite
510   cz    114.14125   $ holtite
511   cz    116.760625  $ holtite
512   cz    119.53875      $ holtite - total 4.1875 inches
513   cz    120.80875      $ outer steel shell - 0.5 inches
c     512   cz    119.38      $ holtite - total 4.125 inches
c     513   cz    120.65      $ outer steel shell - 0.5 inches
514   cz    111.54      $ hole in pocket trunion
515   cz    105.7275    $ flange bottom of overpack
516   cz    105.7275    $ flange top    of overpack
517   cz    109.5375    $ shear ring
518   cz    103.1875    $ item 14  1.25 inch
519   cz    108.2675    $ impact limiter - 2 inch steel
c
521   cz    162.56      $ surface of impact limiters
522   cz    203.1875    $ 1 meter from 506 - upper and lower part overpack
523   cz    220.80875   $ 1 meter from 513 - outer steel
524   cz    303.1875    $ 2 meter from 506 - upper and lower part overpack
525   cz    320.80875   $ 2 meter from 513 - outer steel
526   cz    362.56      $ 2 meter from 521 - edge of impact limiters
527   cz    400.00
c
600   pz    0.0       $ bottom of overpack
601   pz    3.048
602   pz    6.096
603   pz    9.144
604   pz    12.192
610   pz    15.24     $ overpack baseplate   - 6 inches
615   pz    18.415
620   pz    21.59     $ MPC baseplate        - 2.5 inches
630   pz    22.225    $ beginning of item 17 - 0.25 inches
635   pz    23.495    $ item 17              - 0.5 inches
640   pz    41.75125  $ hole in pocket trun  - 7.6875 inches from 630
645   pz    54.61     $ top of pocket trun   - 12.75 inches from 630
660   pz    455.6125  $ top of holtite       - 170.125 inches from 635
665   pz    460.6925  $ top of foam          - 2 inches
670   pz    461.9625  $ top of item 17 on top- 0.5 inches
675   pz    473.71    $ bottom of MPC in lid - 178 inches from 620
676   pz    476.5675  $ top of shear ring
677   pz    494.03    $ top of add steel
651   pz    476.885
652   pz    480.06
653   pz    483.235
654   pz    486.41
655   pz    489.585
656   pz    492.76
657   pz    495.935
680   pz    499.11    $ top of MPC outer lid
685   pz    500.6975  $ bot of overpack lid  - 5/8 inch
686   pz    503.7455
```

```
687   pz    506.7935
688   pz    509.8415
689   pz    512.8895
695   pz    515.9375  $ top of overpack lid  - 6 inches
c
c     tally segment surfaces
c
701   pz    -121.92
702   pz    -91.44
703   pz    -60.96
704   pz    -30.48
c     600   pz    0.0       $ bottom of overpack
c     630   pz    22.225    $ beginning of item 17 - 0.25 inches
705   pz    51.5408
706   pz    80.8567
707   pz    110.1725
708   pz    139.4883
709   pz    168.8042
710   pz    198.1200
711   pz    227.4358
712   pz    256.7517
713   pz    286.0675
714   pz    315.3833
715   pz    344.6992
716   pz    374.0150
717   pz    403.3308
718   pz    432.6467
c     670   pz    461.9625  $ top of item 17 on top- 0.5 inches
719   pz    488.3150
c     695   pz    514.6675  $ top of overpack lid  - 6 inches
720   pz    545.1475
721   pz    575.6275
722   pz    606.1075
723   pz    636.5875
c
740   cz    15.24
741   cz    45.72
742   cz    76.2
743   cz    106.68
744   cz    137.16
745   cz    167.64
746   cz    198.12
747   cz    228.6
748   cz    259.08
749   cz    289.56
750   cz    320.04
751   cz    350.52
752   cz    381.0
c
801   pz    -2.54    $ steel disk
802   pz    -5.715   $ holtite
803   pz    -8.89    $ holtite
804   pz    -9.2075  $ cover over holtite
805   pz    -53.34   $ item 2 on impact limiters
810   pz    -100.0   $ 1 meter from surface overpack
811   pz    -105.7275 $ edge of impact limiter
812   pz    -200.0    $ 2 meter from surface overpack
813   pz    -305.7275 $ 2 meter from surface impact limiter
814   pz    -427.6475 $ 2 meter + 4 feet
815   pz    -488.6075 $ 2 meter + 6 feet
816   pz    -671.4875 $ 2 meter + 12 feet
```

```
817   pz    -700.00
c
821   pz    518.4775    $ steel disk
822   pz    521.6525    $ holtite
823   pz    524.8275    $ holtite
824   pz    525.145     $ cover over holtite
825   pz    569.2775    $ item 2 on impact limiters
830   pz    615.9375    $ 1 meter from surface overpack
831   pz    621.6650    $ edge of impact limiter
832   pz    715.9375    $ 2 meter from surface overpack
833   pz    821.6650    $ 2 meter from surface impact limiter
834   pz    943.5850    $ 2 meter + 4 feet
835   pz    1004.545    $ 2 meter + 6 feet
836   pz    1187.425    $ 2 meter + 12 feet
837   pz    1200.00
c
c         steel spine and holtite cells
c
1000      px    0.0
1001      px  -0.635
1002      px   0.635
1011   1  px  -0.635
1012   1  px   0.635
1021   2  px  -0.635
1022   2  px   0.635
1031   3  px  -0.635
1032   3  px   0.635
1041   4  px  -0.635
1042   4  px   0.635
1051   5  px  -0.635
1052   5  px   0.635
1061   6  px  -0.635
1062   6  px   0.635
1071   7  px  -0.635
1072   7  px   0.635
1081   8  px  -0.635
1082   8  px   0.635
1091   9  px  -0.635
1092   9  px   0.635
c
1101   px   15.71625  $ pocket trunion
1102   px   8.09625   $ pocket trunion opening
1103   px  -8.09625   $ pocket trunion opening 6 3/8 inches thick
1104   px  -15.71625  $ pocket trunion - 9 3/8 inches thick
c
2000      py   0.0
2001      py  -0.635
2002      py   0.635
2011   1  py  -0.635
2012   1  py   0.635
2021   2  py  -0.635
2022   2  py   0.635
2031   3  py  -0.635
2032   3  py   0.635
2041   4  py  -0.635
2042   4  py   0.635
2051   5  py  -0.635
2052   5  py   0.635
2061   6  py  -0.635
2062   6  py   0.635
2071   7  py  -0.635
```

```
2072   7  py   0.635
2081   8  py  -0.635
2082   8  py   0.635
2091   9  py  -0.635
2092   9  py   0.635
c
2101   py   15.71625  $ pocket trunion
2102   py   8.09625   $ pocket trunion opening
2103   py  -8.09625   $ pocket trunion opening 6 3/8 inches thick
2104   py  -15.71625  $ pocket trunion - 9 3/8 inches thick
c
c     OVERPACK surfaces /\ /\ /\ /\ /\
c
c     BLANK LINE

c     BLANK LINE
c
*tr1    0 0 0  9 279 90  99  9 90 90 90 0
*tr2    0 0 0 18 288 90 108 18 90 90 90 0
*tr3    0 0 0 27 297 90 117 27 90 90 90 0
*tr4    0 0 0 36 306 90 126 36 90 90 90 0
*tr5    0 0 0 45 315 90 135 45 90 90 90 0
*tr6    0 0 0 54 324 90 144 54 90 90 90 0
*tr7    0 0 0 63 333 90 153 63 90 90 90 0
*tr8    0 0 0 72 342 90 162 72 90 90 90 0
*tr9    0 0 0 81 351 90 171 81 90 90 90 0
c
c         PHOTON MATERIALS
c
c     fuel 3.4 w/o U235  10.412 gm/cc
c     m1           92235.01p   -0.029971
c                  92238.01p   -0.851529
c                  8016.01p    -0.1185
c     c     homogenized fuel  density 3.979996 gm/cc
c     m2           92235.01p   -0.024483
c                  92238.01p   -0.695601
c                   8016.01p   -0.096801
c                  40000.01p   -0.183115
c     c     zirconium 6.55 gm/cc
c     m3           40000.01p   1.           $ Zr Clad
c     c     stainless steel 7.92 gm/cc
c     m5           24000.01p   -0.19
c                  25055.01p   -0.02
c                  26000.01p   -0.695
c                  28000.01p   -0.095
c     c     boral  2.644 gm/cc
c     m6           5010.01p    -0.044226
c                  5011.01p    -0.201474
c                  13027.01p   -0.6861
c                  6000.01p    -0.0682
c     c     holtite 1.61 gm/cc
c     m7           6000.01p  -0.2766039
c                 13027.01p  -0.21285
c                  1001.01p  -0.0592
c                  8016.01p  -0.42372
c                  7014.01p  -0.0198
c                  5010.01p  -0.0014087
c                  5011.01p  -0.0064174
c     c     carbon steel 7.82 gm/cc
c     m8      6000.01p -0.005 26000.01p -0.995
c     c     air density 1.17e-3 gm/cc
```

```
c     m9      7014.01p 0.78 8016.01p 0.22
c     c
c     c         NEUTRON MATERIALS
c
c     fuel 3.4 w/o U235  10.412 gm/cc
m1           92235.50c   -0.029971
             92238.50c   -0.851529
             8016.50c    -0.1185
c     homogenized fuel  density 3.979996 gm/cc
m2           92235.50c   -0.024483
             92238.50c   -0.695601
              8016.50c   -0.096801
             40000.35c   -0.183115
c     helium  1e-4 gm/cc
m3     2004.50c 1.0
c     stainless steel 7.92 gm/cc
m5           24000.50c   -0.19
             25055.50c   -0.02
             26000.55c   -0.695
             28000.50c   -0.095
c     boral  2.644 gm/cc
m6           5010.50c    -0.044226
             5011.56c    -0.201474
             13027.50c   -0.6861
             6000.50c    -0.0682
c     holtite 1.61 gm/cc
m7           6000.50c  -0.2766039
            13027.50c  -0.21285
             1001.50c  -0.0592
             8016.50c  -0.42372
             7014.50c  -0.0198
             5010.50c  -0.0014087
             5011.56c  -0.0064174
mt7    lwtr.01t
c     carbon steel 7.82 gm/cc
m8      6000.50c -0.005 26000.55c -0.995
c     air density 1.17e-3 gm/cc
m9      7014.50c 0.78 8016.50c 0.22
c
phys:n   20 0.0
phys:p   100 0
c     imp:n   1 228r 0
c     imp:p   1 228r 0
nps     500000
prdmp    j   -30   1   2
c     print   10 110 160 161 20 170
print
mode n p
c
sdef  par=1  erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
c    energy dist for gammas in the fuel
c
c    si1  h  0.7 1.0 1.5 2.0 2.5 3.0
c    sp1     0  0.31 0.31 0.15 0.15 0.08
c
c    energy dist for neutrons in the fuel
c
si1  h  0.1 0.4 0.9 1.4 1.85 3.0 6.43 20.0
sp1     0   0.03787 0.1935 0.1773 0.1310 0.2320 0.2098 0.01853
c
```

```
c    energy dist for Co60 gammas
c
c    si1  d 1.3325 1.1732
c    sp1    0.5    0.5
c
c    axial dist for neut and phot in fuel
c
si3  h 40.3479 55.5879 70.8279 101.3079 162.2679 223.2279
       284.1879 345.1479 375.6279 390.8679 406.1079
sp3    0 0.00005 0.00961 0.07031 0.23323 0.25719 0.22907
         0.16330 0.03309 0.00409 0.00005
sb3    0 1 1 1 1 1 1 1 1 1 1
c
c    axial dist for Co60 - a zero prob is in the fuel
c
c    si3  h 21.59 40.3479 421.3479 445.4271 448.8561 457.3397 468.63
c    sp3    0     0.547   0.0      0.125    0.045    0.227    0.056
c
si4  s               15 16
               13 14 15 16 17 18
            12 13 14 15 16 17 18 19
            12 13 14 15 16 17 18 19
         11 12 13 14 15 16 17 18 19 20
         11 12 13 14 15 16 17 18 19 20
            12 13 14 15 16 17 18 19
            12 13 14 15 16 17 18 19
               13 14 15 16 17 18
                     15 16
sp4  1 67r
c
ds5  s               30 30
               29 29 29 29 29 29
            28 28 28 28 28 28 28 28
            27 27 27 27 27 27 27 27
         26 26 26 26 26 26 26 26 26 26
         25 25 25 25 25 25 25 25 25 25
            24 24 24 24 24 24 24 24
            23 23 23 23 23 23 23 23
               22 22 22 22 22 22
                     21 21
c
si11  -80.6831 -67.6783
si12  -64.1985 -51.1937
si13  -47.7139 -34.7091
si14  -31.2293 -18.2245
si15  -14.7447 -1.7399
si16   1.7399   14.7447
si17   18.2245  31.2293
si18   34.7091  47.7139
si19   51.1937  64.1985
si20   67.6783  80.6831
c
si21  -80.6831 -67.6783
si22  -64.1985 -51.1937
si23  -47.7139 -34.7091
si24  -31.2293 -18.2245
si25  -14.7447 -1.7399
si26   1.7399   14.7447
si27   18.2245  31.2293
si28   34.7091  47.7139
si29   51.1937  64.1985
```

```
si30   67.6783  80.6831
sp11  0 1
sp12  0 1
sp13  0 1
sp14  0 1
sp15  0 1
sp16  0 1
sp17  0 1
sp18  0 1
sp19  0 1
sp20  0 1
sp21  0 1
sp22  0 1
sp23  0 1
sp24  0 1
sp25  0 1
sp26  0 1
sp27  0 1
sp28  0 1
sp29  0 1
sp30  0 1
c
#        imp:n    imp:p
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   307      2        1
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   314      2        1
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c
c       neutron dose factors
c
c    2.5e-8   1.0e-7   1.0e-6   1.0e-5   1.0e-4   1.0e-3   1.0e-2     0.1
c        0.5      1.0      2.5      5.0      7.0      10.0      14.0     20.0
c    3.67e-6  3.67e-6  4.46e-6  4.54e-6  4.18e-6  3.76e-6  3.56e-6 2.17e-5
c        9.26e-5  1.32e-4  1.25e-4  1.56e-4  1.47e-4  1.47e-4  2.08e-4 2.27e-4
c
c       photon dose factors
c
c    0.01  0.03  0.05  0.07  0.1  0.15  0.2  0.25  0.3  0.35  0.4  0.45
c         0.5  0.55  0.6  0.65  0.7  0.8  1.0  1.4  1.8  2.2  2.6  2.8  3.25
c         3.75  4.25  4.75  5.0  5.25  5.75  6.25  6.75  7.5  9.0  11.0
c         13.0  15.0
c   3.96e-06  5.82e-07  2.90e-07  2.58e-07  2.83e-07  3.79e-07  5.01e-07
c        6.31e-07  7.59e-07  8.78e-07  9.85e-07  1.08e-06  1.17e-06  1.27e-06
c        1.36e-06  1.44e-06  1.52e-06  1.68e-06  1.98e-06  2.51e-06  2.99e-06
c        3.42e-06  3.82e-06  4.01e-06  4.41e-06  4.83e-06  5.23e-06  5.60e-06
c        5.80e-06  6.01e-06  6.37e-06  6.74e-06  7.11e-06  7.66e-06  8.77e-06
c        1.03e-05  1.18e-05  1.33e-05
c
f2:n   600 810 811 812 813
fs2    -740 -741 -742 -743 -744 -745 -746 -747 -748 -749 -750 -751 -752 t
fc2    1ft all
ft2    scx 1
de2   2.5e-8   1.0e-7   1.0e-6   1.0e-5   1.0e-4   1.0e-3   1.0e-2     0.1
      0.5      1.0      2.5      5.0      7.0      10.0      14.0     20.0
df2   3.67e-6  3.67e-6  4.46e-6  4.54e-6  4.18e-6  3.76e-6  3.56e-6 2.17e-5
      9.26e-5  1.32e-4  1.25e-4  1.56e-4  1.47e-4  1.47e-4  2.08e-4 2.27e-4
fq2   u s
```

```
tf2   3j 2
c
f12:n  695 830 831 832 833
fs12   -740 -741 -742 -743 -744 -745 -746 -747 -748 -749 -750 -751 -752 t
fc12    1ft all
ft12    scx 1
de12   2.5e-8   1.0e-7   1.0e-6   1.0e-5   1.0e-4   1.0e-3   1.0e-2     0.1
      0.5      1.0      2.5      5.0      7.0      10.0      14.0     20.0
df12   3.67e-6  3.67e-6  4.46e-6  4.54e-6  4.18e-6  3.76e-6  3.56e-6 2.17e-5
      9.26e-5  1.32e-4  1.25e-4  1.56e-4  1.47e-4  1.47e-4  2.08e-4 2.27e-4
fq12   u s
tf12   3j 2
c
c      PHOTON TALLIES
c
f102:p   600 810 811 812 813
fs102    -740 -741 -742 -743 -744 -745 -746 -747 -748 -749 -750 -751 -752 t
fc102    1ft all
ft102    scx 1
de102    0.01  0.03  0.05  0.07  0.1  0.15  0.2  0.25  0.3  0.35  0.4  0.45
         0.5  0.55  0.6  0.65  0.7  0.8  1.0  1.4  1.8  2.2  2.6  2.8  3.25
         3.75  4.25  4.75  5.0  5.25  5.75  6.25  6.75  7.5  9.0  11.0
         13.0  15.0
df102   3.96e-06  5.82e-07  2.90e-07  2.58e-07  2.83e-07  3.79e-07  5.01e-07
        6.31e-07  7.59e-07  8.78e-07  9.85e-07  1.08e-06  1.17e-06  1.27e-06
        1.36e-06  1.44e-06  1.52e-06  1.68e-06  1.98e-06  2.51e-06  2.99e-06
        3.42e-06  3.82e-06  4.01e-06  4.41e-06  4.83e-06  5.23e-06  5.60e-06
        5.80e-06  6.01e-06  6.37e-06  6.74e-06  7.11e-06  7.66e-06  8.77e-06
        1.03e-05  1.18e-05  1.33e-05
fq102   u s
tf102   3j 2
c
f112:p   695 830 831 832 833
fs112    -740 -741 -742 -743 -744 -745 -746 -747 -748 -749 -750 -751 -752 t
fc112    1ft all
ft112    scx 1
de112    0.01  0.03  0.05  0.07  0.1  0.15  0.2  0.25  0.3  0.35  0.4  0.45
         0.5  0.55  0.6  0.65  0.7  0.8  1.0  1.4  1.8  2.2  2.6  2.8  3.25
         3.75  4.25  4.75  5.0  5.25  5.75  6.25  6.75  7.5  9.0  11.0
         13.0  15.0
df112   3.96e-06  5.82e-07  2.90e-07  2.58e-07  2.83e-07  3.79e-07  5.01e-07
        6.31e-07  7.59e-07  8.78e-07  9.85e-07  1.08e-06  1.17e-06  1.27e-06
        1.36e-06  1.44e-06  1.52e-06  1.68e-06  1.98e-06  2.51e-06  2.99e-06
        3.42e-06  3.82e-06  4.01e-06  4.41e-06  4.83e-06  5.23e-06  5.60e-06
        5.80e-06  6.01e-06  6.37e-06  6.74e-06  7.11e-06  7.66e-06  8.77e-06
        1.03e-05  1.18e-05  1.33e-05
fq112   u s
tf112   3j 2
```

CHAPTER 6: CRITICALITY EVALUATION

This chapter documents the criticality evaluation of the HI-STAR 100 System for the storage of spent nuclear fuel in accordance with 10CFR72.124. The results of this evaluation demonstrate that the HI-STAR 100 System is consistent with the Standard Review Plan for Dry Cask Storage Systems, NUREG-1536, and thus, fulfills the following acceptance criteria:

1. The multiplication factor (k_{eff}), including all biases and uncertainties at a 95-percent confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.

2. At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, should occur before an accidental criticality is deemed to be possible.

3. When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both.

4. Criticality safety of the cask system should not rely on use of the following credits:

    a. burnup of the fuel
    b. fuel-related burnable neutron absorbers
    c. more than 75 percent for fixed neutron absorbers when subject to standard acceptance test†.

In addition to demonstrating that the criticality safety acceptance criteria are satisfied, this chapter describes the HI-STAR 100 System design structures and components important to criticality safety and defines the limiting fuel characteristics.

Much of the new information in this chapter is directly extracted from previously NRC approved Holtec dockets; this information is shown in italics. In Chapter 6, this information was extracted from Reference [6.0.1]. All changes in this revision are marked with revision bars*.

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† *For greater credit allowance, fabrication tests capable of verifying the presence and uniformity of the neutron absorber are needed.*

* Wherever multiple units are shown, the US units are the governing value, and the SI units are shown for information only.

6.1 DISCUSSION AND RESULTS

In conformance with the principles established in NUREG-1536 [6.1.1], 10CFR72.124 [6.1.2], and NUREG-0800 Section 9.1.2 [6.1.3], the results in this chapter demonstrate that the effective multiplication factor (k_{eff}) of the HI-STAR 100 System, including all biases and uncertainties evaluated with a 95% probability at the 95% confidence level, does not exceed 0.95 under all credible normal, off-normal, and accident conditions. Moreover, these results demonstrate that the HI-STAR 100 System is designed and maintained such that at least two unlikely, independent, and concurrent or sequential changes must occur to the conditions essential to criticality safety before a nuclear criticality accident is possible. These criteria provide a large subcritical margin, sufficient to assure the criticality safety of the HI-STAR 100 System when fully loaded with fuel of the highest permissible reactivity.

Criticality safety of the HI-STAR 100 System depends on the following principal design parameters:

1. The inherent geometry of the fuel basket designs within the MPC (and the flux-trap water gaps in the MPC-24);

2. The incorporation of permanent fixed neutron-absorbing panels in the fuel basket structure;

3. An administrative limit on the maximum enrichment for PWR fuel and maximum planar-average enrichment for BWR fuel; and

4. *An administrative limit on the minimum soluble boron concentration in the water for loading/unloading fuel with higher enrichments in the MPC-24, and for loading/unloading fuel in the MPC-32.*

The normal conditions for loading/unloading, handling, packaging, transfer, and storage of the HI-STAR 100 System conservatively include: full flooding with ordinary water corresponding to the highest reactivity, and the worst case (most conservative) combination of manufacturing and fabrication tolerances. The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 11 have no adverse effect on the design parameters important to criticality safety, and thus, the off-normal and accident conditions are identical to those for normal conditions.

The HI-STAR 100 System is designed such that the fixed neutron absorber will remain effective for a storage period greater than 20 years, and there are no credible means to lose it. Therefore, in accordance with 10CFR72.124(b), there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

Criticality safety of the HI-STAR 100 System does not rely on the use of any of the following credits:

- burnup of fuel
- fuel-related burnable neutron absorbers
- more than 75 percent of the B-10 content for the Boral fixed neutron absorber .
- *more than 90 percent of the B-10 content for the Metamic fixed neutron absorber, with comprehensive fabrication tests as described in Section 9.1.5.3.2.*

The following interchangeable basket designs are available for use in the HI-STAR 100 System:

- a 24-cell basket (MPC-24), designed for intact PWR fuel assemblies with a specified maximum enrichment *and, for higher enrichments, a minimum soluble boron concentration in the pool water for loading/unloading operations.*
- a 68-cell basket (MPC-68), designed for both intact and damaged BWR fuel assemblies with a specified maximum planar-average enrichment. Additionally, a variation in the MPC-68, designated MPC-68F, is designed for damaged BWR fuel assemblies and BWR fuel debris with a specified maximum planar-average enrichment.
- *a 32-cell basket (MPC-32), designed for intact PWR fuel assemblies of a specified maximum enrichment and minimum soluble boron concentration for loading/unloading.*

Two interchangeable neutron absorber materials are used in these baskets, Boral and Metamic. For Boral, 75 percent of the minimum B-10 content is credited in the criticality analysis, while for Metamic, 90 percent of the minimum B-10 content is credited, based on the neutron absorber tests specified in Section 9.1.5.3. However, the B-10 content in Metamic is chosen to be lower than the B-10 content in Boral, and is chosen so that the absolute B-10 content credited in the criticality analysis is the same for the two materials. This makes the two materials identical from a criticality perspective. This is confirmed by comparing results for a selected number of cases that were performed with both materials (see Section 6.4.10). Calculations in this chapter are therefore only performed for the Boral neutron absorber, with results directly applicable to Metamic.

The HI-STAR 100 System for storage is dry (no moderator), and thus, the reactivity is very low (k_{eff} < *0.52*). However, the HI-STAR 100 System for loading and unloading operations is flooded, and thus, represents the limiting case in terms of reactivity.

Confirmation of the criticality safety of the HI-STAR 100 System under flooded conditions, when filled with fuel of the maximum permissible reactivity for which they are designed, was accomplished with the three-dimensional Monte Carlo code MCNP4a [6.1.4]. Independent confirmatory calculations were made with NITAWL-KENO5a from the SCALE-4.3 package. KENO5a [6.1.5] calculations used the 238-group SCALE cross-section library compiled with the NITAWL-II program [6.1.6], which adjusts the uranium-238 cross sections to compensate for resonance self-shielding effects. The Dancoff factors required by NITAWL-II were calculated with the CELLDAN code [6.1.13], which includes the SUPERDAN code [6.1.7] as a subroutine. K-factors for one-sided statistical tolerance limits with 95% probability at the 95% confidence level were obtained from the National Bureau of Standards (now NIST) Handbook 91 [6.1.8].

CASMO-3, a two-dimensional transport theory code [6.1.9-6.1.12] for fuel assemblies, was used to assess the incremental reactivity effects due to manufacturing tolerances. The CASMO-3 calculations identify those tolerances that cause a positive reactivity effect, enabling the Monte Carlo code input to define the worst case (most conservative) conditions. CASMO-3 was not used for quantitative information, but only to qualitatively indicate the direction and approximate magnitude of the reactivity effects of the manufacturing tolerances.

Benchmark calculations were made to compare the primary code packages (MCNP4a and KENO5a) with experimental data, using critical experiments selected to encompass, insofar as practical, the design parameters of the HI-STAR 100 System. The most important parameters are (1) the enrichment, (2) the water-gap size (MPC-24) or cell spacing (*MPC-32 and* MPC-68), (3) the 10B loading of the neutron absorber panels, *and (4) the soluble boron concentration in the water (MPC-24 and MPC-32).* Benchmark calculations are presented in Appendix 6.A.

Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, USNRC, Washington D.C., January 1997.

- 10CFR72.124, Criteria For Nuclear Criticality Safety.

- Code of Federal Regulations, Title 10, Part 50, Appendix A, General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.

- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 3, July 1981.

To assure the true reactivity will always be less than the calculated reactivity, the following conservative assumptions were made:

- The MPCs are assumed to contain the most reactive fresh fuel authorized to be loaded into a specific basket design.

- Consistent with NUREG-1536, no credit for fuel burnup is assumed, either in depleting the quantity of fissile nuclides or in producing fission product poisons.

- Consistent with NUREG-1536, the criticality analyses assume 75% of the manufacturer's minimum Boron-10 content for the Boral neutron absorber *and 90% of the manufacturer's minimum Boron-10 content for the Metamic neutron absorber*.

- The fuel stack density is conservatively assumed to be 96% of theoretical (10.522 g/cm^3) for all criticality analyses. No credit is taken for fuel pellet dishing or chamfering.

- No credit is taken for the ^{234}U and ^{236}U in the fuel.

- When flooded, the moderator is assumed to be pure, unborated water at a temperature corresponding to the highest reactivity within the expected operating range (i.e., water density of 1.000 g/cc).

- *When credit is taken for soluble boron, a ^{10}B content of 18.0 wt% in boron is assumed.*

- Neutron absorption in minor structural members and heat conduction elements is neglected, i.e., spacer grids, basket supports, and aluminum heat conduction elements are replaced by water.

- Consistent with NUREG-1536, the worst hypothetical combination of tolerances (most conservative values within the range of acceptable values), as identified in Section 6.3, is assumed.

- When flooded, the fuel rod pellet-to-clad gap regions are assumed to be flooded.

- Planar-averaged enrichments are assumed for BWR fuel. (In accordance with NUREG-1536, analysis is presented in Appendix 6.B to demonstrate that the use of planar-average enrichments produces conservative results.)

- Consistent with NUREG-1536, fuel-related burnable neutron absorbers, such as the Gadolinia normally used in BWR fuel and IFBA normally used in PWR fuel, are neglected.

- Higher temperatures of the fuel and moderator resulting from decay heat are neglected.

- For evaluation of the bias, all benchmark calculations that result in a k_{eff} greater than 1.0 are conservatively truncated to 1.0000, consistent with NUREG-1536.

- The water reflector above and below the fuel is assumed to be unborated water.

- For fuel assemblies that contain low-enriched axial blankets, the governing enrichment is that of the highest planar average, and the blankets are not included in determining the average enrichment.

- *Regarding the position of assemblies in the basket, configurations with centered and* *eccentric positioning of assemblies in the fuel storage locations are considered. For further discussions see Section 6.3.3.*

- For intact fuel assemblies, as defined in Appendix B to the Certificate of Compliance, missing fuel rods are assumed to be replaced with dummy rods that displace an amount of water greater than or equal to the original rods.

Results of the design basis criticality safety calculations for single unreflected, internally flooded casks (limiting cases) are listed in Tables 6.1.1 through 6.1.6, conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.3), and including the calculational bias, uncertainties, and calculational statistics. For each of the MPC designs, *minimum soluble boron concentration (if applicable)* and fuel assembly classes†, Tables 6.1.1 through 6.1.6 list the bounding maximum k_{eff} value and the associated maximum allowable enrichment. The maximum allowed enrichments are defined in Appendix B to the Certificate of Compliance. Maximum k_{eff} values for each of the candidate fuel assemblies and basket configurations, that are bounded by those listed in Tables 6.1.1 through 6.1.6, are given in Section 6.2.

A table listing the maximum k_{eff} (including bias, uncertainties, and calculational statistics), calculated k_{eff}, standard deviation, and energy of the average lethargy causing fission (EALF) for each of the candidate fuel assemblies and basket configurations is provided in Appendix 6.C. These results confirm that the maximum k_{eff} values for the HI-STAR 100 System are below the limiting design criteria ($k_{eff} < 0.95$) when fully flooded and loaded with any of the candidate fuel assemblies and basket configurations. Analyses for the various conditions of flooding that support the conclusion that the fully flooded condition corresponds to the highest reactivity, and thus is most limiting, are presented in Section 6.4. The capability of the HI-STAR 100 System to safely accommodate damaged fuel and fuel debris is demonstrated in Subsection 6.4.4.

Accident conditions have also been considered and no credible accident has been identified that would result in exceeding the design criteria limit on reactivity. After the MPC is loaded with spent fuel, it is seal-welded and cannot be internally flooded. The HI-STAR 100 System for storage is dry (no moderator) and the reactivity is very low. For arrays of HI-STAR 100 casks, the radiation shielding and the physical separation between overpacks due to the large diameter and cask pitch preclude any significant neutronic coupling between the casks.

† For each array size (e.g., 6x6, 7x7, 14x14, etc.), the fuel assemblies have been subdivided into a number of assembly classes, where an assembly class is defined in terms of the (1) number of fuel rods; (2) pitch; (3) number and location of guide tubes (PWR) or water rods (BWR); and (4) cladding material. The assembly classes for BWR and PWR fuel are defined in Section 6.2.

Table 6.1.1

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24
(no soluble boron)

Fuel Assembly Class	**Maximum Allowable Enrichment (wt% ^{235}U)**	**Maximum† k_{eff}**
14x14A	4.6	0.9296
14x14B	4.6	0.9228
14x14C	4.6	0.9287
14x14D	4.0	0.8507
15x15A	4.1	0.9204
15x15B	4.1	0.9388
15x15C	4.1	0.9361
15x15D	4.1	0.9367
15x15E	4.1	0.9368
15x15F	4.1	0.9395††
15x15G	4.0	0.8876
15x15H	3.8	0.9337
16x16A	4.6	0.9287
17x17A	4.0	0.9368
17x17B	4.0	0.9324
17x17C	4.0	0.9336

Note: These calculations are for single unreflected, fully flooded casks. However, comparable reactivities were obtained for fully reflected casks and for arrays of casks.

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

†† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9378.

Table 6.1.2

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68

Fuel Assembly Class	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**	**Maximum† k_{eff}**
6x6A	2.7††	0.7888†††
6x6B‡	2.7††	0.7824†††
6x6C	2.7††	0.8021†††
7x7A	2.7††	0.7974†††
7x7B	4.2	0.9386
8x8A	2.7††	0.7697†††
8x8B	4.2	0.9416
8x8C	4.2	0.9425
8x8D	4.2	0.9403
8x8E	4.2	0.9312
8x8F	3.6	0.9153

Note: These calculations are for single unreflected, fully flooded casks. However, comparable reactivities were obtained for fully reflected casks and for arrays of casks.

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

†† This calculation was performed for 3.0% planar-average enrichment, however, the actual fuel is limited, as specified in Appendix B to the CoC, to a maximum planar-average enrichment of 2.7%. Therefore, the listed maximum k_{eff} value is conservative.

††† This calculation was performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68 is 0.0372 g/cm^2. Therefore, the listed maximum k_{eff} value is conservative.

‡ Assemblies in this class contain both MOX and UO_2 pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is given in the Appendix B to the Certificate of Compliance.

Table 6.1.2 (continued)

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum† k_{eff}
9x9A	4.2	0.9417
9x9B	4.2	0.9436
9x9C	4.2	0.9395
9x9D	4.2	0.9394
9x9E	4.1	0.9424
9x9F	4.1	0.9424
10x10A	4.2	0.9457††
10x10B	4.2	0.9436
10x10C	4.2	0.9433
10x10D	4.0	0.9376
10x10E	4.0	0.9185

Note: These calculations are for single unreflected, fully flooded casks. However, comparable reactivities were obtained for fully reflected casks and for arrays of casks.

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

†† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9453.

Table 6.1.3

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68F

Fuel Assembly Class	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**	**Maximum† k_{eff}**
6x6A	2.7††	0.7888
6x6B†††	2.7	0.7824
6x6C	2.7	0.8021
7x7A	2.7	0.7974
8x8A	2.7	0.7697

Note:

1. These calculations are for single unreflected, fully flooded casks. However, comparable reactivities were obtained for fully reflected casks and for arrays of casks.

2. These calculations were performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68F is 0.010 g/cm^2. Therefore, the listed maximum k_{eff} values are conservative.

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

†† These calculations were performed for 3.0% planar-average enrichment, however, the actual fuel is limited, as specified in Appendix B to the CoC, to a maximum planar-average enrichment of 2.7%. Therefore, the listed maximum k_{eff} values are conservative.

††† Assemblies in this class contain both MOX and UO_2 pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is specified in Appendix B to the Certificate of Compliance.

Table 6.1.4

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24 WITH 400 PPM SOLUBLE BORON

Fuel Assembly Class	***Maximum Allowable Enrichment (wt% ^{235}U)***	***Maximum† k_{eff}***
14x14A	*5.0*	*0.8884*
14x14B	*5.0*	*0.8900*
14x14C	*5.0*	*0.8950*
14x14D	*5.0*	*0.8518*
15x15A	*5.0*	*0.9119*
15x15B	*5.0*	*0.9284*
15x15C	*5.0*	*0.9236*
15x15D	*5.0*	*0.9261*
15x15E	*5.0*	*0.9265*
15x15F	*5.0*	*0.9314*
15x15G	*5.0*	*0.8939*
15x15H	*5.0*	*0.9366*
16x16A	*5.0*	*0.8993*
17x17A	*5.0*	*0.9264*
17x17B	*5.0*	*0.9284*
17x17C	*5.0*	*0.9294*

Note: The results are for unreflected, internally fully flooded HI-STAR casks.

† *The term "maximum k_{eff}" as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.*

Table 6.1.5

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-32 FOR 4.1% ENRICHMENT

Fuel Assembly Class	***Maximum Allowable Enrichment (wt% ^{235}U)***	***Minimum Soluble Boron Concentration (ppm)*** *	***Maximum*** † ***k_{eff}***
14x14A	*4.1*	*1300*	*0.9041*
14x14B	*4.1*	*1300*	*0.9257*
14x14C	*4.1*	*1300*	*0.9423*
14x14D	*4.1*	*1300*	*0.8970*
15x15A	*4.1*	*1800*	*0.9206*
15x15B	*4.1*	*1800*	*0.9397*
15x15C	*4.1*	*1800*	*0.9266*
15x15D	*4.1*	*1900*	*0.9384*
15x15E	*4.1*	*1900*	*0.9365*
15x15F	*4.1*	*1900*	*0.9411*
15x15G	*4.1*	*1800*	*0.9147*
15x15H	*4.1*	*1900*	*0.9276*
16x16A	*4.1*	*1400*	*0.9375*
17x17A	*4.1*	*1900*	*0.9111*
17x17B	*4.1*	*1900*	*0.9309*
17x17C	*4.1*	*1900*	*0.9355*

Note: The results are for unreflected, internally fully flooded HI-STAR casks.

* *For maximum allowable enrichments between 4.1 wt% ^{235}U and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified in Table 6.1.5 and Table 6.1.6 for each assembly class.*

† *The term "maximum k_{eff}" as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.*

Table 6.1.6

BOUNDING MAXIMUM keff VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-32 FOR 5.0% ENRICHMENT

Fuel Assembly Class	***Maximum Allowable Enrichment (wt% ^{235}U)***	***Minimum Soluble Boron Concentration (ppm)****	***Maximum† k_{eff}***
14x14A	*5.0*	*1900*	*0.9000*
14x14B	*5.0*	*1900*	*0.9214*
14x14C	*5.0*	*1900*	*0.9480*
14x14D	*5.0*	*1900*	*0.9050*
15x15A	*5.0*	*2500*	*0.9230*
15x15B	*5.0*	*2500*	*0.9429*
15x15C	*5.0*	*2500*	*0.9307*
15x15D	*5.0*	*2600*	*0.9466*
15x15E	*5.0*	*2600*	*0.9434*
15x15F	*5.0*	*2600*	*0.9483*
15x15G	*5.0*	*2500*	*0.9251*
15x15H	*5.0*	*2600*	*0.9333*
16x16A	*5.0*	*2000*	*0.9429*
17x17A	*5.0*	*2600*	*0.9161*
17x17B	*5.0*	*2600*	*0.9371*
17x17C	*5.0*	*2600*	*0.9437*

Note: The results are for unreflected, internally fully flooded HI-STAR casks.

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* *For maximum allowable enrichments between 4.1 wt% ^{235}U and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified in Table 6.1.5 and Table 6.1.6 for each assembly class.*

† *The term "maximum k_{eff}" as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.*

6.2 SPENT FUEL LOADING

Specifications for the BWR and PWR fuel assemblies that were analyzed in this criticality evaluation are given in Tables 6.2.1 and 6.2.2, respectively. For the BWR fuel characteristics, the number and dimensions for the water rods are the actual number and dimensions. For the PWR fuel characteristics, the actual number and dimensions of the control rod guide tubes and thimbles are used. Table 6.2.1 lists 56 unique BWR assemblies while Table 6.2.2 lists 41 unique PWR assemblies, all of which were explicitly analyzed for this evaluation. Examination of Tables 6.2.1 and 6.2.2 reveals that there are a large number of minor variations in fuel assembly dimensions.

Due to the large number of minor variations in the fuel assembly dimensions, the use of explicit dimensions in the Certificate of Compliance could limit the applicability of the HI-STAR 100 System. To resolve this limitation, bounding criticality analyses are presented in this section for a number of defined fuel assembly classes for both fuel types (PWR and BWR). The results of the bounding criticality analyses justify using bounding specifications for fuel dimensions in the Certificate of Compliance.

6.2.1 Definition of Assembly Classes

For each array size (e.g., 6x6, 7x7, 15x15, etc.), the fuel assemblies have been subdivided into a number of defined classes, where a class is defined in terms of (1) the number of fuel rods; (2) pitch; (3) number and locations of guide tubes (PWR) or water rods (BWR); and (4) cladding material. The assembly classes for BWR and PWR fuel are defined in Tables 6.2.1 and 6.2.2, respectively. It should be noted that these assembly classes are unique to this evaluation and are not known to be consistent with any class designations in the open literature.

For each assembly class, calculations have been performed for all of the dimensional variations for which data is available (i.e., all data in Tables 6.2.1 and 6.2.2). These calculations demonstrate that the maximum reactivity corresponds to:

- maximum active fuel length,
- maximum fuel pellet diameter,
- minimum cladding outside diameter (OD),
- maximum cladding inside diameter (ID),
- minimum guide tube/water rod thickness, and
- maximum channel thickness (for BWR assemblies only).

Therefore, for each assembly class, a bounding assembly was defined based on the above characteristics and a calculation for the bounding assembly was performed to demonstrate compliance with the regulatory requirement of $k_{eff} < 0.95$. In some assembly classes this bounding assembly corresponds directly to one of the actual (real) assemblies; while in most assembly classes, the bounding assembly is artificial (i.e., based on bounding dimensions from more than one of the actual assemblies). In classes where the bounding assembly is artificial, the reactivity of the actual (real) assemblies is typically much less than that of the bounding assembly; thereby providing additional conservatism. As a result of these analyses, the Certificate of Compliance will define acceptability in terms of the bounding assembly parameters for each class.

To demonstrate that the aforementioned characteristics are bounding, a parametric study was performed for a reference BWR assembly, designated herein as 8x8C04 (identified generally as a GE8x8R*). Additionally, parametric studies were performed for a PWR assembly (the 15x15F assembly class) in the MPC-24 and MPC-32 with soluble boron in the water flooding the MPC.* The results of these studies are shown in Table 6.2.3, and verify the positive reactivity effect associated with (1) increasing the pellet diameter, (2) maximizing the cladding ID (while maintaining a constant cladding OD), (3) minimizing the cladding OD (while maintaining a constant cladding ID), (4) decreasing the water rod thickness, (5) artificially replacing the Zircaloy water rod tubes with water, and (6) maximizing the channel thickness. These results, and the many that follow, justify the approach for using bounding dimensions in the Certificate of Compliance. Where margins permit, the Zircaloy water rod tubes (BWR assemblies) are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the Certificate of Compliance. *As these studies were performed with and without soluble boron, they also demonstrate that the bounding dimensions are valid independent of the soluble boron concentration.*

As mentioned, the bounding approach used in these analyses often results in a maximum k_{eff} value for a given class of assemblies that is much greater than the reactivity of any of the actual (real) assemblies within the class, and yet, is still below the 0.95 regulatory limit.

6.2.2 PWR Fuel Assemblies

6.2.2.1 PWR Fuel Assemblies in the MPC-24 without Soluble Boron

For PWR fuel assemblies (specifications listed in Table 6.2.2) the 15x15F01 fuel assembly at 4.1% enrichment has the highest reactivity (maximum k_{eff} of 0.9395). The 17x17A01 assembly (otherwise known as a Westinghouse 17x17 OFA) has a similar reactivity (see Table 6.2.17) and was used throughout this criticality evaluation as a reference PWR assembly. The 17x17A01 assembly is a representative PWR fuel assembly in terms of design and reactivity and is useful for the reactivity studies presented in Sections 6.3 and 6.4. Calculations for the various PWR fuel assemblies in the MPC-24 are summarized in Tables 6.2.4 through 6.2.19 for the fully flooded condition *without soluble boron in the water*.

Tables 6.2.4 through 6.2.19 show the maximum k_{eff} values for the assembly classes that are acceptable for storage in the MPC-24. All maximum k_{eff} values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. All calculations for the MPC-24 were performed for a ^{10}B loading of 0.020 g/cm^2, which is 75% of the minimum loading, 0.0267 g/cm^2, specified on the MPC-24 drawing in Section 1.5. The maximum allowable enrichment in the MPC-24 varies from 4.0 to 4.6 wt% ^{235}U, depending on the assembly class, and is defined in Tables 6.2.4 through 6.2.19. It should be noted that the maximum allowable enrichment does not vary within an assembly class. Table 6.1.1 summarizes the maximum allowable enrichments for each of the assembly classes that are acceptable for storage in the MPC-24.

Tables 6.2.4 through 6.2.19 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and k_{eff} values in the following rows above the bold double lines, and the bounding dimensions selected for the Certificate of Compliance and corresponding bounding k_{eff} values in the final rows. Where the bounding assembly corresponds directly to one of the actual assemblies, the fuel assembly designation is listed in the bottom row in parentheses (e.g., Table 6.2.4). Otherwise, the bounding assembly is given a unique designation. For an assembly class that contains only a single assembly (e.g., 14x14D, see Table 6.2.7), the Certificate of Compliance dimensions are based on the assembly dimensions from that single assembly. All of the maximum k_{eff} values corresponding to the selected bounding dimensions are greater than or equal to those for the actual assembly dimensions and are below the 0.95 regulatory limit.

The results of the analyses for the MPC-24, which were performed for all assemblies in each class (see Tables 6.2.4 through 6.2.19), further confirm the validity of the bounding dimensions established in Section 6.2.1. Thus, for all following calculations, namely analyses of the MPC-32 and MPC-24 with soluble boron present in the water, only the bounding assembly in each class is analyzed.

6.2.2.2 PWR Fuel Assemblies in the MPC-24 with Soluble Boron

Additionally, the HI-STAR 100 system is designed to allow credit for the soluble boron typically present in the water of PWR spent fuel pools. For a minimum soluble boron concentration of 400 ppm, the maximum allowable fuel enrichment is 5.0 wt% ^{235}U for all assembly classes identified in Tables 6.2.4 through 6.2.19. Table 6.1.4 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit. The 15x15H assembly class has the highest reactivity (maximum k_{eff} of 0.9366). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

6.2.2.3 PWR Assemblies in the MPC-32

When loading any PWR fuel assembly in the MPC-32, a minimum soluble boron concentration is required.

For a maximum allowable fuel enrichment of 4.1 wt% ^{235}U for all assembly classes identified in Tables 6.2.4 through 6.2.19, a minimum soluble boron concentration between 1300 ppm and 1900 ppm is required, depending on the assembly class. Table 6.1.5 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 *regulatory limit. The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.*

For a maximum allowable fuel enrichment of 5.0 wt% ^{235}U for all assembly classes identified in Tables 6.2.4 through 6.2.19, a minimum soluble boron concentration between 1900 ppm and 2600 ppm is required, depending on the assembly class. Table 6.1.6 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit. The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

It is desirable to limit the soluble boron concentration to a level appropriate for the maximum enrichment in a basket, since this prevents adding soluble boron unnecessarily to the spent fuel pool during loading and unloading operations. This approach requires a minimum soluble boron level as a function of the maximum allowable enrichment, which can be directly derived by linear interpolation from the calculations at 4.1 wt% ^{235}U and 5.0 wt% ^{235}U shown in Tables 6.1.5 and 6.1.6. Since the maximum k_{eff} is a near linear function of both enrichment and soluble boron concentration, linear interpolation is both appropriate and sufficient. Further, studies have shown that this approach results in maximum k_{eff} values for enrichments between 4.1 wt% ^{235}U and 5.0 wt% ^{235}U that are lower than those maximum k_{eff} values calculated at 4.1 wt% and 5.0 wt% ^{235}U in Tables 6.1.5 and 6.1.6.

6.2.3 BWR Fuel Assemblies in the MPC-68

For BWR fuel assemblies (specifications listed in Table 6.2.1) the artificial bounding assembly for the 10x10A assembly class at 4.2% enrichment has the highest reactivity (maximum k_{eff} of 0.9457). Calculations for the various BWR fuel assemblies in the MPC-68 are summarized in Tables 6.2.20 through 6.2.36 for the fully flooded condition. In all cases, the gadolinia (Gd_2O_3) normally incorporated in BWR fuel was conservatively neglected.

For calculations involving BWR assemblies, the use of a uniform (planar-average) enrichment, as opposed to the distributed enrichments normally used in BWR fuel, produces conservative results. Calculations confirming this statement are presented in Appendix 6.B for several representative BWR fuel assembly designs. These calculations justify the specification of planar-average enrichments to define acceptability of BWR fuel for loading into the MPC-68.

Tables 6.2.20 through 6.2.36 show the maximum k_{eff} values for assembly classes that are acceptable for storage in the MPC-68. All maximum k_{eff} values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. With the exception of assembly classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A, which will be discussed in Section 6.2.4, all calculations for the MPC-68 were performed with a ^{10}B loading of 0.0279 g/cm^2, which is 75% of the minimum loading, 0.0372 g/cm^2, specified on the MPC-68 drawing in Section 1.5. Calculations for assembly classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A were conservatively performed with a ^{10}B loading of 0.0067 g/cm^2. The maximum allowable enrichment in the MPC-68 varies from 2.7 to 4.2 wt% ^{235}U, depending on the assembly class. It should be noted that the maximum allowable enrichment does not vary within an assembly class. Table 6.1.2 summarizes the maximum allowable enrichments for all assembly classes that are acceptable for storage in the MPC-68.

Tables 6.2.20 through 6.2.36 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and k_{eff} values in the following rows above the bold double lines, and the bounding dimensions selected for the Certificate of Compliance and corresponding bounding k_{eff} values in the final rows. Where an assembly class contains only a single assembly (e.g., 8x8E, see Table 6.2.24), the Certificate of Compliance dimensions are based on the assembly dimensions from that single assembly. For assembly classes that are suspected to contain assemblies with thicker channels (e.g., 120 mils or 0.3048 cm), bounding calculations are also performed to qualify the thicker channels (e.g. 7x7B, see Table 6.2.20). All of the maximum k_{eff} values corresponding to the selected bounding dimensions are shown to be greater than or equal to those for the actual assembly dimensions and are below the 0.95 regulatory limit.

For assembly classes that contain partial length rods (i.e., 9x9A, 10x10A, and 10x10B), calculations were performed for the actual (real) assembly configuration and for the axial segments (assumed to be full length) with and without the partial length rods. In all cases, the axial segment with only the full length rods present (where the partial length rods are absent) is bounding. Therefore, the bounding maximum k_{eff} values reported for assembly classes that contain partial length rods bound the reactivity regardless of the active fuel length of the partial length rods. As a result, the Certificate of Compliance have no minimum requirement for the active fuel length of the partial length rods.

For BWR fuel assembly classes where margins permit, the Zircaloy water rod tubes are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the Certificate of Compliance. For these cases, the bounding water rod thickness is listed as zero.

As mentioned, the highest observed maximum k_{eff} value is 0.9457, corresponding to the artificial bounding assembly in the 10x10A assembly class. This assembly has the following bounding characteristics: (1) the partial length rods are assumed to be zero length (most reactive configuration); (2) the channel is assumed to be 120 mils (0.3048 cm) thick; and (3) the active fuel length of the full length rods is 155 inches (393.7 cm). Therefore, the maximum reactivity value is bounding compared to any of the real BWR assemblies listed.

6.2.4 Damaged BWR Fuel Assemblies and BWR Fuel Debris

In addition to storing intact PWR and BWR fuel assemblies, the HI-STAR 100 System is designed to store damaged BWR fuel assemblies and BWR fuel debris. Damaged fuel assemblies and fuel debris are defined in Section 2.1.3 and Appendix B to the Certificate of Compliance. Both damaged BWR fuel assemblies and BWR fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. Two different DFC types with slightly different cross sections are considered. DFCs containing fuel debris must be stored in the MPC-68F. DFCs containing damaged fuel assemblies may be stored in either the MPC-68 or MPC-68F. The criticality evaluation of various possible damaged conditions of the fuel is presented in Subsection 6.4.4 for both DFC types.

Tables 6.2.37 through 6.2.41 show the maximum k_{eff} values for the six assembly classes that may be stored as damaged fuel or fuel debris. All maximum k_{eff} values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. All calculations were performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum loading, 0.0089 g/cm^2. However, because the practical manufacturing lower limit for minimum ^{10}B loading is 0.01 g/cm^2, the minimum ^{10}B loading of 0.01 g/cm^2 is specified on the MPC-68 drawing in Section 1.5, for the MPC-68F. As an additional level of conservatism in the analyses, the calculations were performed for an enrichment of 3.0 wt% ^{235}U, while the maximum allowable enrichment for these assembly classes is limited to 2.7 wt% ^{235}U in the Certificate of Compliance. Therefore, the maximum k_{eff} values for damaged BWR fuel assemblies and fuel debris are conservative. Calculations for the various BWR fuel assemblies in the MPC-68F are summarized in Tables 6.2.37 through 6.2.41 for the fully flooded condition.

For the assemblies that may be stored as damaged fuel or fuel debris, the 6x6C01 assembly at 3.0 wt% ^{235}U enrichment has the highest reactivity (maximum k_{eff} of 0.8021). Considering all of the conservatism built into this analysis (e.g., higher than allowed enrichment and lower than actual ^{10}B loading), the actual reactivity will be lower.

Because the analysis for the damaged BWR fuel assemblies and fuel debris was performed for a ^{10}B loading of 0.0089 g/cm^2, which conservatively bounds damaged BWR fuel assemblies in a standard MPC-68 with a minimum ^{10}B loading of 0.0372 g/cm^2, damaged BWR fuel assemblies may also be stored in the standard MPC-68. However, fuel debris is limited to the MPC-68F by Appendix B to the Certificate of Compliance.

Tables 6.2.37 through 6.2.41 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and k_{eff} values in the following rows above the bold double lines, and the bounding dimensions selected for the Certificate of Compliance and corresponding bounding k_{eff} values in the final rows. Where an assembly class contains only a single assembly (e.g., 6x6C, see Table 6.2.39), the Certificate of Compliance dimensions are based on the assembly dimensions from that single assembly. All of the maximum k_{eff} values corresponding to the selected bounding dimensions are greater than or equal to those for the actual assembly dimensions and are well below the 0.95 regulatory limit.

6.2.5 Thoria Rod Canister

Additionally, the HI-STAR 100 System is designed to store a Thoria Rod Canister in the MPC-68 or MPC-68F. The canister is similar to a DFC and contains 18 intact Thoria Rods placed in a separator assembly. The reactivity of the canister in the MPC-68 or MPC-68F is very low compared to the reactivity of the approved fuel assemblies (The ^{235}U content of these rods corresponds to UO_2 rods with an initial enrichment of approximately 1.7 wt% ^{235}U). It is therefore permissible to store the Thoria Rod Canister together with any other approved content in a MPC-68 or MPC-68F. Specifications of the canister and the Thoria Rods that are used in the criticality evaluation are given in Table 6.2.42. The criticality evaluation is presented in Subsection 6.4.6.

Table 6.2.1 (page 1 of 6)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
6x6A Assembly Class												
6x6A01	Zr	0.694 (1.763)	36	0.5645 (1.4338)	0.0350 (0.0889)	0.4940 (1.2548)	110.0 (279.4)	0	n/a	n/a	0.060 (0.152)	4.290 (10.897)
6x6A02	Zr	0.694 (1.763)	36	0.5645 (1.4338)	0.0360 (0.0914)	0.4820 (1.2243)	110.0 (279.4)	0	n/a	n/a	0.060 (0.152)	4.290 (10.897)
6x6A03	Zr	0.694 (1.763)	36	0.5645 (1.4338)	0.0350 (0.0889)	0.4820 (1.2243)	110.0 (279.4)	0	n/a	n/a	0.060 (0.152)	4.290 (10.897)
6x6A04	Zr	0.694 (1.763)	36	0.5550 (1.4097)	0.0350 (0.0889)	0.4820 (1.2243)	110.0 (279.4)	0	n/a	n/a	0.060 (0.152)	4.290 (10.897)
6x6A05	Zr	0.696 (1.768)	36	0.5625 (1.4288)	0.0350 (0.0889)	0.4820 (1.2243)	110.0 (279.4)	0	n/a	n/a	0.060 (0.152)	4.290 (10.897)
6x6A06	Zr	0.696 (1.768)	35	0.5625 (1.4288)	0.0350 (0.0889)	0.4820 (1.2243)	110.0 (279.4)	1	0.0 (0.0)	0.0 (0.0)	0.060 (0.152)	4.290 (10.897)
6x6A07	Zr	0.700 (1.778)	36	0.5555 (1.4110)	0.03525 (0.0895)	0.4780 (1.2141)	110.0 (279.4)	0	n/a	n/a	0.060 (0.152)	4.290 (10.897)
6x6A08	Zr	0.710 (1.803)	36	0.5625 (1.4288)	0.0260 (0.0660)	0.4980 (1.2649)	110.0 (279.4)	0	n/a	n/a	0.060 (0.152)	4.290 (10.897)
6x6B (MOX) Assembly Class												
6x6B01	Zr	0.694 (1.763)	36	0.5645 (1.4338)	0.0350 (0.0889)	0.4820 (1.2243)	110.0 (279.4)	0	n/a	n/a	0.060 (0.152)	4.290 (10.897)
6x6B02	Zr	0.694 (1.763)	36	0.5625 (1.4288)	0.0350 (0.0889)	0.4820 (1.2243)	110.0 (279.4)	0	n/a	n/a	0.060 (0.152)	4.290 (10.897)
6x6B03	Zr	0.696 (1.768)	36	0.5625 (1.4288)	0.0350 (0.0889)	0.4820 (1.2243)	110.0 (279.4)	0	n/a	n/a	0.060 (0.152)	4.290 (10.897)
6x6B04	Zr	0.696 (1.768)	35	0.5625 (1.4288)	0.0350 (0.0889)	0.4820 (1.2243)	110.0 (279.4)	1	0.0 (0.0)	0.0 (0.0)	0.060 (0.152)	4.290 (10.897)
6x6B05	Zr	0.710 (1.803)	35	0.5625 (1.4288)	0.0350 (0.0889)	0.4820 (1.2243)	110.0 (279.4)	1	0.0 (0.0)	0.0 (0.0)	0.060 (0.152)	4.290 (10.897)
6x6C Assembly Class												
6x6C01	Zr	0.740 (1.880)	36	0.5630 (1.4300)	0.0320 (0.0813)	0.4880 (1.2395)	77.5 (196.85)	0	n/a	n/a	0.060 (0.152)	4.542 (11.537)
7x7A Assembly Class												
7x7A01	Zr	0.631 (1.603)	49	0.4860 (1.2344)	0.0328 (0.0833)	0.4110 (1.0439)	80 (203.2)	0	n/a	n/a	0.060 (0.152)	4.542 (11.537)

Table 6.2.1 (page 2 of 6)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
7x7B Assembly Class												
7x7B01	Zr	0.738 (1.875)	49	0.5630 (1.4300)	0.0320 (0.0813)	0.4870 (1.2370)	150 (381)	0	n/a	n/a	0.080 (0.203)	5.278 (13.406)
7x7B02	Zr	0.738 (1.875)	49	0.5630 (1.4300)	0.0370 (0.0940)	0.4770 (1.2116)	150 (381)	0	n/a	n/a	0.102 (0.259)	5.291 (13.439)
7x7B03	Zr	0.738 (1.875)	49	0.5630 (1.4300)	0.0370 (0.0940)	0.4770 (1.2116)	150 (381)	0	n/a	n/a	0.080 (0.203)	5.278 (13.406)
7x7B04	Zr	0.738 (1.875)	49	0.5700 (1.4478)	0.0355 (0.0902)	0.4880 (1.2395)	150 (381)	0	n/a	n/a	0.080 (0.203)	5.278 (13.406)
7x7B05	Zr	0.738 (1.875)	49	0.5630 (1.4300)	0.0340 (0.0864)	0.4775 (1.2129)	150 (381)	0	n/a	n/a	0.080 (0.203)	5.278 (13.406)
7x7B06	Zr	0.738 (1.875)	49	0.5700 (1.4478)	0.0355 (0.0902)	0.4910 (1.2471)	150 (381)	0	n/a	n/a	0.080 (0.203)	5.278 (13.406)
8x8A Assembly Class												
8x8A01	Zr	0.523 (1.328)	64	0.4120 (1.0465)	0.0250 (0.0635)	0.3580 (0.9093)	110 (279.4)	0	n/a	n/a	0.100 (0.254)	4.290 (10.897)
8x8A02	Zr	0.523 (1.328)	63	0.4120 (1.0465)	0.0250 (0.0635)	0.3580 (0.9093)	120 (304.8)	0	n/a	n/a	0.100 (0.254)	4.290 (10.897)

Table 6.2.1 (page 3 of 6)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
8x8B Assembly Class												
8x8B01	Zr	0.641 (1.628)	63	0.4840 (1.2294)	0.0350 (0.0889)	0.4050 (1.0287)	150 (381)	1	0.484 (1.229)	0.414 (1.052)	0.100 (0.254)	5.278 (13.406)
8x8B02	Zr	0.636 (1.615)	63	0.4840 (1.2294)	0.0350 (0.0889)	0.4050 (1.0287)	150 (381)	1	0.484 (1.229)	0.414 (1.052)	0.100 (0.254)	5.278 (13.406)
8x8B03	Zr	0.640 (1.626)	63	0.4930 (1.2522)	0.0340 (0.0864)	0.4160 (1.0566)	150 (381)	1	0.493 (1.252)	0.425 (1.080)	0.100 (0.254)	5.278 (13.406)
8x8B04	Zr	0.642 (1.631)	64	0.5015 (1.2738)	0.0360 (0.0914)	0.4195 (1.0655)	150 (381)	0	n/a	n/a	0.100 (0.254)	5.278 (13.406)
8x8C Assembly Class												
8x8C01	Zr	0.641 (1.628)	62	0.4840 (1.2294)	0.0350 (0.0889)	0.4050 (1.0287)	150 (381)	2	0.484 (1.229)	0.414 (1.052)	0.100 (0.254)	5.278 (13.406)
8x8C02	Zr	0.640 (1.626)	62	0.4830 (1.2268)	0.0320 (0.0813)	0.4100 (1.0414)	150 (381)	2	0.591 (1.501)	0.531 (1.349)	0.000 (0.000)	no channel
8x8C03	Zr	0.640 (1.626)	62	0.4830 (1.2268)	0.0320 (0.0813)	0.4100 (1.0414)	150 (381)	2	0.591 (1.501)	0.531 (1.349)	0.080 (0.203)	5.278 (13.406)
8x8C04	Zr	0.640 (1.626)	62	0.4830 (1.2268)	0.0320 (0.0813)	0.4100 (1.0414)	150 (381)	2	0.591 (1.501)	0.531 (1.349)	0.100 (0.254)	5.278 (13.406)
8x8C05	Zr	0.640 (1.626)	62	0.4830 (1.2268)	0.0320 (0.0813)	0.4100 (1.0414)	150 (381)	2	0.591 (1.501)	0.531 (1.349)	0.120 (0.305)	5.278 (13.406)
8x8C06	Zr	0.640 (1.626)	62	0.4830 (1.2268)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	2	0.591 (1.501)	0.531 (1.349)	0.100 (0.254)	5.278 (13.406)
8x8C07	Zr	0.640 (1.626)	62	0.4830 (1.2268)	0.0340 (0.0864)	0.4100 (1.0414)	150 (381)	2	0.591 (1.501)	0.531 (1.349)	0.100 (0.254)	5.278 (13.406)
8x8C08	Zr	0.640 (1.626)	62	0.4830 (1.2268)	0.0320 (0.0813)	0.4100 (1.0414)	150 (381)	2	0.493 (1.252)	0.425 (1.080)	0.100 (0.254)	5.278 (13.406)
8x8C09	Zr	0.640 (1.626)	62	0.4930 (1.2522)	0.0340 (0.0864)	0.4160 (1.0566)	150 (381)	2	0.493 (1.252)	0.425 (1.080)	0.100 (0.254)	5.278 (13.406)
8x8C10	Zr	0.640 (1.626)	62	0.4830 (1.2268)	0.0340 (0.0864)	0.4100 (1.0414)	150 (381)	2	0.591 (1.501)	0.531 (1.349)	0.120 (0.305)	5.278 (13.406)
8x8C11	Zr	0.640 (1.626)	62	0.4830 (1.2268)	0.0340 (0.0864)	0.4100 (1.0414)	150 (381)	2	0.591 (1.501)	0.531 (1.349)	0.120 (0.305)	5.215 (13.246)
8x8C12	Zr	0.636 (1.615)	62	0.4830 (1.2268)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	2	0.591 (1.501)	0.531 (1.349)	0.120 (0.305)	5.215 (13.246)

Table 6.2.1 (page 4 of 6)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
8x8D Assembly Class												
8x8D01	Zr	0.640 (1.626)	60	0.4830 (1.2268)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	2 large/ 2 small	0.591/ 0.483 (1.501/1.227)	0.531/ 0.433 (1.349/1.100)	0.100 (0.254)	5.278 (13.406)
8x8D02	Zr	0.640 (1.626)	60	0.4830 (1.2268)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	4	0.591 (1.501)	0.531 (1.349)	0.100 (0.254)	5.278 (13.406)
8x8D03	Zr	0.640 (1.626)	60	0.4830 (1.2268)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	4	0.483 (1.227)	0.433 (1.100)	0.100 (0.254)	5.278 (13.406)
8x8D04	Zr	0.640 (1.626)	60	0.4830 (1.2268)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	1	1.34 (3.404)	1.26 (3.200)	0.100 (0.254)	5.278 (13.406)
8x8D05	Zr	0.640 (1.626)	60	0.4830 (1.2268)	0.0320 (0.0813)	0.4100 (1.0414)	150 (381)	1	1.34 (3.404)	1.26 (3.200)	0.100 (0.254)	5.278 (13.406)
8x8D06	Zr	0.640 (1.626)	60	0.4830 (1.2268)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	1	1.34 (3.404)	1.26 (3.200)	0.120 (0.305)	5.278 (13.406)
8x8D07	Zr	0.640 (1.626)	60	0.4830 (1.2268)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	1	1.34 (3.404)	1.26 (3.200)	0.080 (0.203)	5.278 (13.406)
8x8D08	Zr	0.640 (1.626)	61	0.4830 (1.2268)	0.0300 (0.0762)	0.4140 (1.0516)	150 (381)	3	0.591 (1.501)	0.531 (1.349)	0.080 (0.203)	5.278 (13.406)
8x8E Assembly Class												
8x8E01	Zr	0.640 (1.626)	59	0.4930 (1.2522)	0.0340 (0.0864)	0.4160 (1.0566)	150 (381)	5	0.493 (1.252)	0.425 (1.080)	0.100 (0.254)	5.278 (13.406)
8x8F Assembly Class												
8x8F01	Zr	0.609 (1.547)	64	0.4576 (1.1623)	0.0290 (0.0737)	0.3913 (0.9939)	150 (381)	4†	0.291 (0.739)†	0.228 (0.579)†	0.055 (0.140)	5.390 (13.691)
9x9A Assembly Class												
9x9A01	Zr	0.566 (1.438)	74	0.4400 (1.1176)	0.0280 (0.0711)	0.3760 (0.9550)	150 (381)	2	0.98 (2.489)	0.92 (2.337)	0.100 (0.254)	5.278 (13.406)
9x9A02	Zr	0.566 (1.438	66	0.4400 (1.1176)	0.0280 (0.0711)	0.3760 (0.9550)	150 (381)	2	0.98(2.489)	0.92 (2.337)	0.100 (0.254)	5.278 (13.406)
9x9A03	Zr	0.566 (1.438	74/66	0.4400 (1.1176)	0.0280 (0.0711)	0.3760 (0.9550)	150/90 (381/228.6)	2	0.98 (2.489)	0.92 (2.337)	0.100 (0.254)	5.278 (13.406)
9x9A04	Zr	0.566 (1.438	74/66	0.4400 (1.1176)	0.0280 (0.0711)	0.3760 (0.9550)	150/90 (381/228.6)	2	0.98(2.489)	0.92 (2.337)	0.120 (0.305)	5.278 (13.406)

† Four rectangular water cross segments dividing the assembly into four quadrants

Table 6.2.1 (page 5 of 6)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness	Fuel Assembly Designation
9x9B Assembly Class												
9x9B01	Zr	0.569 (1.445)	72	0.4330 (1.0998)	0.0262 (0.0665)	0.3737 (0.9492)	150 (381)	1	1.516 (3.851)	1.459 (3.706)	0.100 (0.254)	5.278 (13.406)
9x9B02	Zr	0.569 (1.445)	72	0.4330 (1.0998)	0.0260 (0.0660)	0.3737 (0.9492)	150 (381)	1	1.516 (3.851)	1.459 (3.706)	0.100 (0.254)	5.278 (13.406)
9x9B03	Zr	0.572 (1.453)	72	0.4330 (1.0998)	0.0260 (0.0660)	0.3737 (0.9492)	150 (381)	1	1.516 (3.851)	1.459 (3.706)	0.100 (0.254)	5.278 (13.406)
9x9C Assembly Class												
9x9C01	Zr	0.572 (1.453)	80	0.4230 (1.0744)	0.0295 (0.0749)	0.3565 (0.9055)	150 (381)	1	0.512 (1.300)	0.472 (1.199)	0.100 (0.254)	5.278 (13.406)
9x9D Assembly Class												
9x9D01	Zr	0.572 (1.453)	79	0.4240 (1.0770)	0.0300 (0.0762)	0.3565 (0.9055)	150 (381)	2	0.424 (1.077)	0.364 (0.925)	0.100 (0.254)	5.278 (13.406)
9x9E Assembly Class†												
9x9E01	Zr	0.572 (1.453)	76	0.4170 (1.0592)	0.0265 (0.0673)	0.3530 (0.8966)	150 (381)	5	0.546 (1.387)	0.522 (1.326)	0.120 (0.305)	5.215 (13.246)
9x9E02	Zr	0.572 (1.453)	48 28	0.4170 (1.0592) 0.4430 (1.1252)	0.0265 (0.0673) 0.0285 (0.0724)	0.3530 (0.8966) 0.3745 (0.9512)	150 (381)	5	0.546 (1.387)	0.522 (1.326)	0.120 (0.305)	5.215 (13.246)
9x9F Assembly Class†												
9x9F01	Zr	0.572 (1.453)	76	0.4430 (1.1252)	0.0285 (0.0724)	0.3745 (0.9512)	150 (381)	5	0.546 (1.387)	0.522 (1.326)	0.120 (0.305)	5.215 (13.246)
9x9F02	Zr	0.572 (1.453)	48 28	0.4170 (1.0592) 0.4430 (1.1252)	0.0265 (0.0673) 0.0285 (0.0724)	0.3530 (0.8966) 0.3745 (0.9512)	150 (381)	5	0.546 (1.387)	0.522 (1.326)	0.120 (0.305)	5.215 (13.246)

† The 9x9E and 9x9F fuel assembly classes represent a single fuel type containing fuel rods with different dimensions (SPC 9x9-5). In addition to the actual configuration (9x9E02 and 9x9F02), the 9x9E class contains a hypothetical assembly with only small fuel rods (9x9E01), and the 9x9F class contains a hypothetical assembly with only large rods (9x9F01). This was done in order to simplify the specification of this assembly in the CoC.

Table 6.2.1 (page 6 of 6)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
10x10A Assembly Class												
10x10A01	Zr	0.510 (1.295)	92	0.4040 (1.0262)	0.0260 (0.0660)	0.3450 (0.8763)	155 (393.7)	2	0.980 (2.489)	0.920 (2.337)	0.100 (0.254)	5.278 (13.406)
10x10A02	Zr	0.510 (1.295)	78	0.4040 (1.0262)	0.0260 (0.0660)	0.3450 (0.8763)	155(393.7)	2	0.980 (2.489)	0.920 (2.337)	0.100 (0.254)	5.278 (13.406)
10x10A03	Zr	0.510 (1.295)	92/78	0.4040 (1.0262)	0.0260 (0.0660)	0.3450 (0.8763)	155/90 (393.7/228.6)	2	0.980 (2.489)	0.920 (2.337)	0.100 (0.254)	5.278 (13.406)
10x10B Assembly Class												
10x10B01	Zr	0.510 (1.295)	91	0.3957 (1.0051)	0.0239 (0.0607)	0.3413 (0.8669)	155(393.7)	1	1.378 (3.500)	1.321 (3.355)	0.100 (0.254)	5.278 (13.406)
10x10B02	Zr	0.510 (1.295)	83	0.3957 (1.0051)	0.0239 (0.0607)	0.3413 (0.8669)	155(393.7)	1	1.378 (3.500)	1.321 (3.355)	0.100 (0.254)	5.278 (13.406)
10x10B03	Zr	0.510 (1.295)	91/83	0.3957 (1.0051)	0.0239 (0.0607)	0.3413 (0.8669)	155/90 (393.7/228.6)	1	1.378 (3.500)	1.321 (3.355)	0.100 (0.254)	5.278 (13.406)
10x10C Assembly Class												
10x10C01	Zr	0.488 (1.240)	96	0.3780 (0.9601)	0.0243 (0.0617)	0.3224 (0.8189)	150(393.7)	5	1.227 (3.117)	1.165 (2.959)	0.055 (0.140)	5.457 (13.861)
10x10D Assembly Class												
10x10D01	SS	0.565 (1.435)	100	0.3960 (1.0058)	0.0200 (0.0508)	0.3500 (0.8890)	83 (210.8)	0	n/a	n/a	0.08 (0.203)	5.663 (14.384)
10x10E Assembly Class												
10x10E01	SS	0.557 (1.415)	96	0.3940 (1.0008)	0.0220 (0.0559)	0.3430 (0.8712)	83 (210.8)	4	0.3940 (1.001)	0.3500 (0.889)	0.08 (0.203)	5.663 (14.384)

Table 6.2.2 (page 1 of 3)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
14x14A Assembly Class											
14x14A01	Zr	0.556 (1.412)	179	0.400 (1.016)	0.0243 (0.0617)	0.3444 (0.8748)	150 (381)	17	0.527 (1.339)	0.493 (1.252)	0.0170 (0.0432)
14x14A02	Zr	0.556 (1.412)	179	0.400 (1.016)	0.0243 (0.0617)	0.3444 (0.8748)	150 (381)	17	0.528 (1.341)	0.490 (1.245)	0.0190 (0.0483)
14x14A03	Zr	0.556 (1.412)	179	0.400 (1.016)	0.0243 (0.0617)	0.3444 (0.8748)	150 (381)	17	0.526 (1.336)	0.492 (1.250)	0.0170 (0.0432)
14x14B Assembly Class											
14x14B01	Zr	0.556 (1.412)	179	0.422 (1.072)	0.0243 (0.0617)	0.3659 (0.9294)	150 (381)	17	0.539 (1.369)	0.505 (1.283)	0.0170 (0.0432)
14x14B02	Zr	0.556 (1.412)	179	0.417 (1.059)	0.0295 (0.0749)	0.3505 (0.8903)	150 (381)	17	0.541 (1.374)	0.507 (1.288)	0.0170 (0.0432)
14x14B03	Zr	0.556 (1.412)	179	0.424 (1.077)	0.0300 (0.0762)	0.3565 (0.9055)	150 (381)	17	0.541 (1.374)	0.507 (1.288)	0.0170 (0.0432)
14x14B04	Zr	0.556 (1.412)	179	0.426 (1.082)	0.0310 (0.0787)	0.3565 (0.9055)	150 (381)	17	0.541 (1.374)	0.507 (1.288)	0.0170 (0.0432)
14x14C Assembly Class											
14x14C01	Zr	0.580 (1.473)	176	0.440 (1.118)	0.0280 (0.0711)	0.3765 (0.9563)	150 (381)	5	1.115 (2.832)	1.035 (2.629)	0.0400 (0.1016)
14x14C02	Zr	0.580 (1.473)	176	0.440 (1.118)	0.0280 (0.0711)	0.3770 (0.9576)	150 (381)	5	1.115 (2.832)	1.035 (2.629)	0.0400 (0.1016)
14x14C03	Zr	0.580 (1.473)	176	0.440 (1.118)	0.0260 (0.0660)	0.3805 (0.9665)	150 (381)	5	1.111 (2.822)	1.035 (2.629)	0.0380 (0.0965)
14x14D Assembly Class											
14x14D01	SS	0.556 (1.412)	180	0.422 (1.072)	0.0165 (0.0419)	0.3835 (0.9741)	144 (365.76)	16	0.543 (1.379)	0.514 (1.306)	0.0145 (0.0368)
15x15A Assembly Class											
15x15A01	Zr	0.550 (1.397)	204	0.418 (1.062)	0.0260 (0.0660)	0.3580 (0.9093)	150 (381)	21	0.533 (1.354)	0.500 (1.270)	0.0165 (0.0419)

Table 6.2.2 (page 2 of 3)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
15x15B Assembly Class											
15x15B01	Zr	0.563 (1.430)	204	0.422 (1.072)	0.0245 (0.0622)	0.3660 (0.9296)	150 (381)	21	0.533 (1.354)	0.499 (1.267)	0.0170 (0.0432)
15x15B02	Zr	0.563 (1.430)	204	0.422 (1.072)	0.0245 (0.0622)	0.3660 (0.9296)	150 (381)	21	0.546 (1.387)	0.512 (1.300)	0.0170 (0.0432)
15x15B03	Zr	0.563 (1.430)	204	0.422 (1.072)	0.0243 (0.0617)	0.3660 (0.9296)	150 (381)	21	0.533 (1.354)	0.499 (1.267)	0.0170 (0.0432)
15x15B04	Zr	0.563 (1.430)	204	0.422 (1.072)	0.0243 (0.0617)	0.3659 (0.9294)	150 (381)	21	0.545 (1.384)	0.515 (1.308)	0.0150 (0.0381)
15x15B05	Zr	0.563 (1.430)	204	0.422 (1.072)	0.0242 (0.0615)	0.3659 (0.9294)	150 (381)	21	0.545 (1.384)	0.515 (1.308)	0.0150 (0.0381)
15x15B06	Zr	0.563 (1.430)	204	0.420 (1.067)	0.0240 (0.0610)	0.3671 (0.9324)	150 (381)	21	0.544 (1.382)	0.514 (1.306)	0.0150 (0.0381)
15x15C Assembly Class											
15x15C01	Zr	0.563 (1.430)	204	0.424 (1.077)	0.0300 (0.0762)	0.3570 (0.9068)	150 (381)	21	0.544 (1.382)	0.493 (1.252)	0.0255 (0.0648)
15x15C02	Zr	0.563 (1.430)	204	0.424 (1.077)	0.0300 (0.0762)	0.3570 (0.9068)	150 (381)	21	0.544 (1.382)	0.511 (1.298)	0.0165 (0.0419)
15x15C03	Zr	0.563 (1.430)	204	0.424 (1.077)	0.0300 (0.0762)	0.3565 (0.9055)	150 (381)	21	0.544 (1.382)	0.511 (1.298)	0.0165 (0.0419)
15x15C04	Zr	0.563 (1.430)	204	0.417 (1.059)	0.0300 (0.0762)	0.3565 (0.9055)	150 (381)	21	0.544 (1.382)	0.511 (1.298)	0.0165 (0.0419)
15x15D Assembly Class											
15x15D01	Zr	0.568 (1.443)	208	0.430 (1.092)	0.0265 (0.0673)	0.3690 (0.9373)	150 (381)	17	0.530 (1.346)	0.498 (1.265)	0.0160 (0.0406)
15x15D02	Zr	0.568 (1.443)	208	0.430 (1.092)	0.0265 (0.0673)	0.3686 (0.9362)	150 (381)	17	0.530 (1.346)	0.498 (1.265)	0.0160 (0.0406)
15x15D03	Zr	0.568 (1.443)	208	0.430 (1.092)	0.0265 (0.0673)	0.3700 (0.9398)	150 (381)	17	0.530 (1.346)	0.499 (1.267)	0.0155 (0.0394)
15x15D04	Zr	0.568 (1.443)	208	0.430 (1.092)	0.0250 (0.0635)	0.3735 (0.9487)	150 (381)	17	0.530 (1.346)	0.500 (1.270)	0.0150 (0.0381)
15x15E Assembly Class											
15x15E01	Zr	0.568 (1.443)	208	0.428 (1.087)	0.0245 (0.0622)	0.3707 (0.9416)	150 (381)	17	0.528 (1.341)	0.500 (1.270)	0.0140 (0.0356)
15x15F Assembly Class											
15x15F01	Zr	0.568 (1.443)	208	0.428 (1.087)	0.0230 (0.0584)	0.3742 (0.9505)	150 (381)	17	0.528 (1.341)	0.500 (1.270)	0.0140 (0.0356)

Table 6.2.2 (page 3 of 3)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
15x15G Assembly Class											
15x15G01	SS	0.563 (1.430)	204	0.422 (1.072)	0.0165 (0.0419)	0.3825 (0.9716)	144 (365.76)	21	0.543 (1.379)	0.514 (1.306)	0.0145 (0.0368)
15x15H Assembly Class											
15x15H01	Zr	0.568 (1.443)	208	0.414 (1.052)	0.0220 (0.0559)	0.3622 (0.9200)	150 (381)	17	0.528 (1.341)	0.500 (1.270)	0.0140 (0.0356)
16x16A Assembly Class											
16x16A01	Zr	0.506 (1.285)	236	0.382 (0.970)	0.0250 (0.0635)	0.3255 (0.8268)	150 (381)	5	0.980 (2.489)	0.900 (2.286)	0.0400 (0.1016)
16x16A02	Zr	0.506 (1.285)	236	0.382 (0.970)	0.0250 (0.0635)	0.3250 (0.8255)	150 (381)	5	0.980 (2.489)	0.900 (2.286)	0.0400 (0.1016)
17x17A Assembly Class											
17x17A01	Zr	0.496 (1.260)	264	0.360 (0.914)	0.0225 (0.0572)	0.3088 (0.7844)	144 (365.76)	25	0.474 (1.204)	0.442 (1.123)	0.0160 (0.0406)
17x17A02	Zr	0.496 (1.260)	264	0.360 (0.914)	0.0225 (0.0572)	0.3088 (0.7844)	150 (381)	25	0.474 (1.204)	0.442 (1.123)	0.0160 (0.0406)
17x17A03	Zr	0.496 (1.260)	264	0.360 (0.914)	0.0250 (0.0635)	0.3030 (0.7696)	150 (381)	25	0.480 (1.219)	0.448 (1.138)	0.0160 (0.0406)
17x17B Assembly Class											
17x17B01	Zr	0.496 (1.260)	264	0.374 (0.950)	0.0225 (0.0572)	0.3225 (0.8192)	150 (381)	25	0.482 (1.224)	0.450 (1.143)	0.0160 (0.0406)
17x17B02	Zr	0.496 (1.260)	264	0.374 (0.950)	0.0225 (0.0572)	0.3225 (0.8192)	150 (381)	25	0.474 (1.204)	0.442 (1.123)	0.0160 (0.0406)
17x17B03	Zr	0.496 (1.260)	264	0.376 (0.955)	0.0240 (0.0610)	0.3215 (0.8166)	150 (381)	25	0.480 (1.219)	0.448 (1.138)	0.0160 (0.0406)
17x17B04	Zr	0.496 (1.260)	264	0.372 (0.945)	0.0205 (0.0521)	0.3232 (0.8209)	150 (381)	25	0.427 (1.085)	0.399 (1.013)	0.0140 (0.0356)
17x17B05	Zr	0.496 (1.260)	264	0.374 (0.950)	0.0240 (0.0610)	0.3195 (0.8115)	150 (381)	25	0.482 (1.224)	0.450 (1.143)	0.0160 (0.0406)
17x17B06	Zr	0.496 (1.260)	264	0.372 (0.945)	0.0205 (0.0521)	0.3232 (0.8209)	150 (381)	25	0.480 (1.219)	0.452 (1.148)	0.0140 (0.0356)
17x17C Assembly Class											
17x17C01	Zr	0.502 (1.275)	264	0.379 (0.963)	0.0240 (0.0610)	0.3232 (0.8209)	150 (381)	25	0.472 (1.199)	0.432 (1.097)	0.0200 (0.0508)
17x17C02	Zr	0.502 (1.275)	264	0.377 (0.958)	0.0220 (0.0559)	0.3252 (0.8260)	150 (381)	25	0.472 (1.199)	0.432 (1.097)	0.0200 (0.0508)

Table 6.2.3(a)
REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS *FOR BWR FUEL IN THE MPC-68*
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	water rod thickness	channel thickness
8x8C04 (GE8x8R)	reference	0.9307	0.0007	0.483 (1.2268)	0.419 (1.0643)	0.032 (0.0813)	0.410 (1.0414)	0.030 (0.0762)	0.100 (0.254)
increase pellet OD (+0.001”)	+0.0005	0.9312	0.0007	0.483 (1.2268)	0.419 (1.0643)	0.032 (0.0813)	0.411 (1.0439)	0.030 (0.0762)	0.100 (0.254)
decrease pellet OD (-0.001”)	-0.0008	0.9299	0.0009	0.483 (1.2268)	0.419 (1.0643)	0.032 (0.0813)	0.409 (1.0389)	0.030 (0.0762)	0.100 (0.254)
increase clad ID (+0.004”)	+0.0027	0.9334	0.0007	0.483 (1.2268)	0.423 (1.0744)	0.030 (0.0762)	0.410 (1.0414)	0.030 (0.0762)	0.100 (0.254)
decrease clad ID (-0.004”)	-0.0034	0.9273	0.0007	0.483 (1.2268)	0.415 (1.0541)	0.034 (0.0864)	0.410 (1.0414)	0.030 (0.0762)	0.100 (0.254)
increase clad OD (+0.004”)	-0.0041	0.9266	0.0008	0.487 (1.2370)	0.419 (1.0643)	0.034 (0.0864)	0.410 (1.0414)	0.030 (0.0762)	0.100 (0.254)
decrease clad OD (-0.004”)	+0.0023	0.9330	0.0007	0.479 (1.2167)	0.419 (1.0643)	0.030 (0.0762)	0.410 (1.0414)	0.030 (0.0762)	0.100 (0.254)
increase water rod thickness (+0.015”)	-0.0019	0.9288	0.0008	0.483 (1.2268)	0.419 (1.0643)	0.032 (0.0813)	0.410 (1.0414)	0.045 (0.1143)	0.100 (0.254)
decrease water rod thickness (-0.015”)	+0.0001	0.9308	0.0008	0.483 (1.2268)	0.419 (1.0643)	0.032 (0.0813)	0.410 (1.0414)	0.015 (0.0381)	0.100 (0.254)
remove water rods (i.e., replace the water rod tubes with water)	+0.0021	0.9328	0.0008	0.483 (1.2268)	0.419 (1.0643)	0.032 (0.0813)	0.410 (1.0414)	0.000 (0.0000)	0.100 (0.254)
remove channel	-0.0039	0.9268	0.0009	0.483 (1.2268)	0.419 (1.0643)	0.032 (0.0813)	0.410 (1.0414)	0.030 (0.0762)	0.000 (0.000)
increase channel thickness (+0.020”)	+0.0005	0.9312	0.0007	0.483 (1.2268)	0.419 (1.0643)	0.032 (0.0813)	0.410 (1.0414)	0.030 (0.0762)	0.120 (0.305)

Table 6.2.3(b)

REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS IN PWR FUEL IN THE MPC 24 WITH 400 PPM SOLUBLE BORON CONCENTRATION

(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated k_{eff}	standard deviation	cladding OD	Cladding ID	cladding thickness	pellet OD	guide tube thickness
15x15F (15x15 B&W, 5.0% E)	reference	0.9271	0.0005	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3742 (0.9505)	0.0140 (0.0356)
increase pellet OD (+0.001”)	-0.0008	0.9263	0.0004	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3752 (0.9530)	0.0140 (0.0356)
decrease pellet OD (-0.001”)	-0.0002	0.9269	0.0005	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3732 (0.9479)	0.0140 (0.0356)
increase clad ID (+0.004”)	+0.0040	0.9311	0.0005	0.4280 (1.0871)	0.3860 (0.9804)	0.0210 (0.0533)	0.3742 (0.9505)	0.0140 (0.0356)
decrease clad ID (-0.004”)	-0.0033	0.9238	0.0004	0.4280 (1.0871)	0.3780 (0.9601)	0.0250 (0.0635)	0.3742 (0.9505)	0.0140 (0.0356)
increase clad OD (+0.004”)	-0.0042	0.9229	0.0004	0.4320 (1.0973)	0.3820 (0.9703)	0.0250 (0.0635)	0.3742 (0.9505)	0.0140 (0.0356)
decrease clad OD (-0.004”)	+0.0035	0.9306	0.0005	0.4240 (1.0770)	0.3820 (0.9703)	0.0210 (0.0533)	0.3742 (0.9505)	0.0140 (0.0356)
increase guide tube thickness (+0.004”)	-0.0008	0.9263	0.0005	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3742 (0.9505)	0.0180 (0.0457)
decrease guide tube thickness (-0.004”)	+0.0006	0.9277	0.0004	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3742 (0.9505)	0.0100 (0.0254)
remove guide tubes (i.e., replace the guide tubes with water)	+0.0028	0.9299	0.0004	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3742 (0.9505)	0.000 (0.000)
voided guide tubes	-0.0318	0.8953	0.0005	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3742 (0.9505)	0.0140 (0.0356)

Table 6.2.3(c)

REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS IN PWR FUEL IN THE MPC-32 WITH 2600 PPM SOLUBLE BORON CONCENTRATION

(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	guide tube thickness
15x15F (15x15 B&W, 5.0% E)	reference	0.9389	0.0004	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3742 (0.9505)	0.0140 (0.0356)
increase pellet OD (+0.001")	+0.0019	0.9408	0.0004	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3752 (0.9530)	0.0140 (0.0356)
decrease pellet OD (-0.001")	0.0000	0.9389	0.0004	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3732 (0.9479)	0.0140 (0.0356)
increase clad ID (+0.004")	+0.0015	0.9404	0.0004	0.4280 (1.0871)	0.3860 (0.9804)	0.0210 (0.0533)	0.3742 (0.9505)	0.0140 (0.0356)
decrease clad ID (-0.004")	-0.0015	0.9374	0.0004	0.4280 (1.0871)	0.3780 (0.9601)	0.0250 (0.0635)	0.3742 (0.9505)	0.0140 (0.0356)
increase clad OD (+0.004")	-0.0002	0.9387	0.0004	0.4320 (1.0973)	0.3820 (0.9703)	0.0250 (0.0635)	0.3742 (0.9505)	0.0140 (0.0356)
decrease clad OD (-0.004")	+0.0007	0.9397	0.0004	0.4240 (1.0770)	0.3820 (0.9703)	0.0210 (0.0533)	0.3742 (0.9505)	0.0140 (0.0356)
increase guide tube thickness (+0.004")	-0.0003	0.9387	0.0004	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3742 (0.9505)	0.0180 (0.0457)
decrease guide tube thickness (-0.004")	-0.0005	0.9384	0.0004	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3742 (0.9505)	0.0100 (0.0254)
remove guide tubes (i.e., replace the guide tubes with water)	-0.0005	0.9385	0.0004	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3742 (0.9505)	0.000 (0.000)
voided guide tubes	+0.0039	0.9428	0.0004	0.4280 (1.0871)	0.3820 (0.9703)	0.0230 (0.0584)	0.3742 (0.9505)	0.0140 (0.0356)

Table 6.2.4
MAXIMUM K_{EFF} VALUES FOR THE 14X14A ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

14x14A (4.6% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 179 fuel rods, 17 guide tubes, pitch=0.556 (1.412), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14A01	0.9295	0.9252	0.0008	0.400 (1.016)	0.3514 (0.8926)	0.0243 (0.0617)	0.3444 (0.8748)	150 (381)	0.017 (0.0432)
14x14A02	0.9286	0.9242	0.0009	0.400 (1.016)	0.3514 (0.8926)	0.0243 (0.0617)	0.3444 (0.8748)	150 (381)	0.019 (0.0483)
14x14A03	0.9296	0.9253	0.0008	0.400 (1.016)	0.3514 (0.8926)	0.0243 (0.0617)	0.3444 (0.8748)	150 (381)	0.017 (0.0432)
Dimensions Listed in Certificate of Compliance				0.400 (1.016) (min.)	0.3514 (0.8926) (max.)		0.3444 (0.8748) (max.)	150 (381) (max.)	0.017 (0.0432) (min.)
bounding dimensions (14x14A03)	0.9296	0.9253	0.0008	0.400 (1.016)	0.3514 (0.8926)	0.0243 (0.0617)	0.3444 (0.8748)	150 (381)	0.017 (0.0432)

Table 6.2.5
MAXIMUM K_{EFF} VALUES FOR THE 14X14B ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

14x14B (4.6% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 179 fuel rods, 17 guide tubes, pitch=0.556 (1.412), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14B01	0.9159	0.9117	0.0007	0.422 (1.072)	0.3734 (0.9484)	0.0243 (0.0617)	0.3659 (0.9294)	150(381)	0.017 (0.0432)
14x14B02	0.9169	0.9126	0.0008	0.417 (1.059)	0.3580 (0.9093)	0.0295 (0.0749)	0.3505 (0.8903)	150(381)	0.017 (0.0432)
14x14B03	0.9110	0.9065	0.0009	0.424 (1.077)	0.3640 (0.9246)	0.0300 (0.0762)	0.3565 (0.9055	150(381)	0.017 (0.0432)
14x14B04	0.9084	0.9039	0.0009	0.426 (1.082)	0.3640 (0.9246)	0.0310 (0.0787)	0.3565 (0.9055	150(381)	0.017 (0.0432)
Dimensions Listed in Certificate of Compliance				0.417 (1.059) (min.)	0.3734 (0.9484) (max.)		0.3659 (0.9294) (max.)	150(381) (max.)	0.017 (0.0432) (min.)
bounding dimensions (B14x14B01)	0.9228	0.9185	0.0008	0.417 (1.059)	0.3734 (0.9484)	0.0218 (0.0554)	0.3659 (0.9294)	150(381)	0.017 (0.0432)

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Table 6.2.6
MAXIMUM K_{EFF} VALUES FOR THE 14X14C ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

14x14C (4.6% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 176 fuel rods, 5 guide tubes, pitch=0.580 (1.473), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14C01	0.9258	0.9215	0.0008	0.440 (1.118)	0.3840 (0.9754)	0.0280 (0.0711)	0.3765 (0.9563)	150(381)	0.040 (0.1016)
14x14C02	0.9265	0.9222	0.0008	0.440 (1.118)	0.3840 (0.9754)	0.0280 (0.0711)	0.3770 (0.9576)	150(381)	0.040 (0.1016)
14x14C03	0.9287	0.9242	0.0009	0.440 (1.118)	0.3880 (0.9855)	0.0260 (0.0660)	0.3805 (0.9665)	150(381)	0.038 (0.0965)
Dimensions Listed in Certificate of Compliance				0.440 (1.118) (min.)	0.3880 (0.9855) (max.)		0.3805 (0.9665) (max.)	150(381) (max.)	0.038 (0.0965) (min.)
bounding dimensions (14x14C01)	0.9287	0.9242	0.0009	0.440 (1.118)	0.3880 (0.9855)	0.0260 (0.0660)	0.3805 (0.9665)	150(381)	0.038 (0.0965)

Table 6.2.7
MAXIMUM K_{EFF} VALUES FOR THE 14X14D ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

14x14D (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 180 fuel rods, 16 guide tubes, pitch=0.556 (1.412), SS clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14D01	0.8507	0.8464	0.0008	0.422 (1.072)	0.3890 (0.9881)	0.0165 (0.0419)	0.3835 (0.9741)	144 (365.76)	0.0145 (0.0368)
Dimensions Listed in Certificate of Compliance				0.422 (1.072) (min.)	0.3890 (0.9881) (max.)		0.3835 (0.9741) (max.)	144 (365.76) (max.)	0.0145 (0.0368) (min.)

Table 6.2.8

MAXIMUM K_{EFF} VALUES FOR THE 15X15A ASSEMBLY CLASS IN THE MPC-24

(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

15x15A (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 204 fuel rods, 21 guide tubes, pitch=0.550 (1.397), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15A01	0.9204	0.9159	0.0009	0.418 (1.062)	0.3660 (0.9296)	0.0260 (0.0660)	0.3580 (0.9093)	150 (381)	0.0165 (0.0419)
Dimensions Listed in Certificate of Compliance				0.418 (1.062) (min.)	0.3660 (0.9296) (max.)		0.3580 (0.9093) (max.)	150 (381) (max.)	0.0165 (0.0419) (min.)

Table 6.2.9
MAXIMUM K_{EFF} VALUES FOR THE 15X15B ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

15x15B (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 204 fuel rods, 21 guide tubes, pitch=0.563 (1.430), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15B01	0.9369	0.9326	0.0008	0.422 (1.072)	0.3730 (0.9474)	0.0245 (0.0622)	0.3660 (0.9296)	150 (381)	0.017 (0.0432)
15x15B02	0.9338	0.9295	0.0008	0.422 (1.072)	0.3730 (0.9474)	0.0245 (0.0622)	0.3660 (0.9296)	150 (381)	0.017 (0.0432)
15x15B03	0.9362	0.9318	0.0008	0.422 (1.072)	0.3734 (0.9484)	0.0243 (0.0617)	0.3660 (0.9296)	150 (381)	0.017 (0.0432)
15x15B04	0.9370	0.9327	0.0008	0.422 (1.072)	0.3734 (0.9484)	0.0243 (0.0617)	0.3659 (0.9294)	150 (381)	0.015 (0.0381)
15x15B05	0.9356	0.9313	0.0008	0.422 (1.072)	0.3736 (0.9489)	0.0242 (0.0615)	0.3659 (0.9294)	150 (381)	0.015 (0.0381)
15x15B06	0.9366	0.9324	0.0007	0.420 (1.067)	0.3720 (0.9449)	0.0240 (0.0610)	0.3671 (0.9324)	150 (381)	0.015 (0.0381)
Dimensions Listed in Certificate of Compliance				0.420 (1.067) (min.)	0.3736 (0.9489) (max.)		0.3671 (0.9324) (max.)	150 (381) (max.)	0.015 (0.0381) (min.)
bounding dimensions (B15x15B01)	0.9388	0.9343	0.0009	0.420 (1.067)	0.3736 (0.9489)	0.0232 (0.0589)	0.3671 (0.9324)	150 (381)	0.015 (0.0381)

Table 6.2.10
MAXIMUM K_{EFF} VALUES FOR THE 15X15C ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

15x15C (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 204 fuel rods, 21 guide tubes, pitch=0.563 (1.430), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15C01	0.9255	0.9213	0.0007	0.424 (1.077)	0.3640 (0.9246)	0.0300 (0.0762)	0.3570 (0.9068)	150 (381)	0.0255 (0.0648)
15x15C02	0.9297	0.9255	0.0007	0.424 (1.077)	0.3640 (0.9246)	0.0300 (0.0762)	0.3570 (0.9068)	150 (381)	0.0165 (0.0419)
15x15C03	0.9297	0.9255	0.0007	0.424 (1.077)	0.3640 (0.9246)	0.0300 (0.0762)	0.3565 (0.9055)	150 (381)	0.0165 (0.0419)
15x15C04	0.9311	0.9268	0.0008	0.417 (1.059)	0.3570 (0.9068)	0.0300 (0.0762)	0.3565 (0.9055)	150 (381)	0.0165 (0.0419)
Dimensions Listed in Certificate of Compliance				0.417 (1.059) (min.)	0.3640 (0.9246) (max.)		0.3570 (0.9068) (max.)	150 (381) (max.)	0.0165 (0.0419) (min.)
bounding dimensions (B15x15C01)	0.9361	0.9316	0.0009	0.417 (1.059)	0.3640 (0.9246)	0.0265 (0.0673)	0.3570 (0.9068)	150 (381)	0.0165 (0.0419)

Table 6.2.11
MAXIMUM K_{EFF} VALUES FOR THE 15X15D ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

15x15D (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 208 fuel rods, 17 guide tubes, pitch=0.568 (1.443), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15D01	0.9341	0.9298	0.0008	0.430 (1.092)	0.3770 (0.9576)	0.0265 (0.0673)	0.3690 (0.9373)	150 (381)	0.0160 (0.0406)
15x15D02	0.9367	0.9324	0.0008	0.430 (1.092)	0.3770 (0.9576)	0.0265 (0.0673)	0.3686 (0.9362)	150 (381)	0.0160 (0.0406)
15x15D03	0.9354	0.9311	0.0008	0.430 (1.092)	0.3770 (0.9576)	0.0265 (0.0673)	0.3700 (0.9398)	150 (381)	0.0155 (0.0394)
15x15D04	0.9339	0.9292	0.0010	0.430 (1.092)	0.3800 (0.9652)	0.0250 (0.0635)	0.3735 (0.9487)	150 (381)	0.0150 (0.0381)
Dimensions Listed in Certificate of Compliance				0.430 (1.092) (min.)	0.3800 (0.9652) (max.)		0.3735 (0.9487) (max.)	150 (381) (max.)	0.0150 (0.0381) (min.)
bounding dimensions (15x15D04)	0.9339†	0.9292	0.0010	0.430 (1.092)	0.3800 (0.9652)	0.0250 (0.0635)	0.3735 (0.9487)	150 (381)	0.0150 (0.0381)

† The k_{eff} value listed for the 15x15D02 case is higher than that for the case with the bounding dimensions. Therefore, the 0.9367 value from case 15x15D02 is listed in Table 6.1.1 as the maximum.

Table 6.2.12
MAXIMUM K_{EFF} VALUES FOR THE 15X15E ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

15x15E (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 208 fuel rods, 17 guide tubes, pitch=0.568 (1.443), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15E01	0.9368	0.9325	0.0008	0.428 (1.087)	0.3790 (0.9627)	0.0245 (0.0622)	0.3707 (0.9416)	150 (381)	0.0140 (0.0356)
Dimensions Listed in Certificate of Compliance				0.428 (1.087) (min.)	0.3790 (0.9627) (max.)		0.3707 (0.9416) (max.)	150 (381) (max.)	0.0140 (0.0356) (min.)

Table 6.2.13
MAXIMUM K_{EFF} VALUES FOR THE 15X15F ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

15x15F (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 208 fuel rods, 17 guide tubes, pitch=0.568 (1.443), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15F01	0.9395†	0.9350	0.0009	0.428 (1.087)	0.3820 (0.9703)	0.0230 (0.0584)	0.3742 (0.9505)	150 (381)	0.0140 (0.0356)
Dimensions Listed in Certificate of Compliance				0.428 (1.087) (min.)	0.3820 (0.9703) (max.)		0.3742 (0.9505) (max.)	150 (381) (max.)	0.0140 (0.0356) (min.)

† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9383.

Table 6.2.14
MAXIMUM K_{EFF} VALUES FOR THE 15X15G ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

15x15G (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 204 fuel rods, 21 guide tubes, pitch=0.563 (1.430), SS clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15G01	0.8876	0.8833	0.0008	0.422 (1.072)	0.3890 (0.9881)	0.0165 (0.0419)	0.3825 (0.9716)	144 (365.76)	0.0145 (0.0368)
Dimensions Listed in Certificate of Compliance				0.422 (1.072) (min.)	0.3890 (0.9881) (max.)		0.3825 (0.9716) (max.)	144 (365.76) (max.)	0.0145 (0.0368) (min.)

Table 6.2.15
MAXIMUM K_{EFF} VALUES FOR THE 15X15H ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

15x15H (3.8% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 208 fuel rods, 17 guide tubes, pitch=0.568 (1.443), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15H01	0.9337	0.9292	0.0009	0.414 (1.052)	0.3700 (0.9398)	0.0220 (0.0559)	0.3622 (0.9200)	150 (381)	0.0140 (0.0356)
Dimensions Listed in Certificate of Compliance				0.414 (1.052) (min.)	0.3700 (0.9398) (max.)		0.3622 (0.9200) (max.)	150 (381) (max.)	0.0140 (0.0356) (min.)

Table 6.2.16
MAXIMUM K_{EFF} VALUES FOR THE 16X16A ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

16x16A (4.6% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 236 fuel rods, 5 guide tubes, pitch=0.506 (1.285), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
16x16A01	0.9287	0.9244	0.0008	0.382 (0.970)	0.3320 (0.8433)	0.0250 (0.0635)	0.3255 (0.8268)	150 (381)	0.0400 (0.1016)
16x16A02	0.9263	0.9221	0.0007	0.382 (0.970)	0.3320 (0.8433)	0.0250 (0.0635)	0.3250 (0.8255)	150 (381)	0.0400 (0.1016)
Dimensions Listed in Certificate of Compliance				0.382 (0.970) (min.)	0.3320 (0.8433) (max.)		0.3255 (0.8268) (max.)	150 (381) (max.)	0.0400 (0.1016) (min.)
bounding dimensions (16x16A01)	0.9287	0.9244	0.0008	0.382 (0.970)	0.3320 (0.8433)	0.0250 (0.0635)	0.3255 (0.8268)	150 (381)	0.0400 (0.1016)

Table 6.2.17
MAXIMUM K_{EFF} VALUES FOR THE 17X17A ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

17x17A (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 264 fuel rods, 25 guide tubes, pitch=0.496 (1.260), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17A01	0.9368	0.9325	0.0008	0.360 (0.914)	0.3150 (0.8001)	0.0225 (0.0572)	0.3088 (0.7844)	144 (365.76)	0.016 (0.0406)
17x17A02	0.9368	0.9325	0.0008	0.360 (0.914)	0.3150 (0.8001)	0.0225 (0.0572)	0.3088 (0.7844)	150 (381)	0.016 (0.0406)
17x17A03	0.9329	0.9286	0.0008	0.360 (0.914)	0.3100 (0.7874)	0.0250 (0.0635)	0.3030 (0.7696)	150 (381)	0.016 (0.0406)
Dimensions Listed in Certificate of Compliance				0.360 (0.914) (min.)	0.3150 (0.8001) (max.)		0.3088 (0.7844) (max.)	150 (381) (max.)	0.016 (0.0406) (min.)
bounding dimensions (17x17A02)	0.9368	0.9325	0.0008	0.360 (0.914)	0.3150 (0.8001)	0.0225 (0.0572)	0.3088 (0.7844)	150 (381)	0.016 (0.0406)

Table 6.2.18
MAXIMUM K_{EFF} VALUES FOR THE 17X17B ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

17x17B (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 264 fuel rods, 25 guide tubes, pitch=0.496 (1.260), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17B01	0.9288	0.9243	0.0009	0.374 (0.950)	0.3290 (0.8357)	0.0225 (0.0572)	0.3225 (0.8192)	150 (381)	0.016 (0.0406)
17x17B02	0.9290	0.9247	0.0008	0.374 (0.950)	0.3290 (0.8357)	0.0225 (0.0572)	0.3225 (0.8192)	150 (381)	0.016 (0.0406)
17x17B03	0.9243	0.9199	0.0008	0.376 (0.955)	0.3280 (0.8331)	0.0240 (0.0610)	0.3215 (0.8166)	150 (381)	0.016 (0.0406)
17x17B04	0.9324	0.9279	0.0009	0.372 (0.945)	0.3310 (0.8407)	0.0205 (0.0521)	0.3232 (0.8209)	150 (381)	0.014 (0.0356)
17x17B05	0.9266	0.9222	0.0008	0.374 (0.950)	0.3260 (0.8280)	0.0240 (0.0610)	0.3195 (0.8115)	150 (381)	0.016 (0.0406)
17x17B06	0.9311	0.9268	0.0008	0.372 (0.945)	0.3310 (0.8407)	0.0205 (0.0521)	0.3232 (0.8209)	150 (381)	0.014 (0.0356)
Dimensions Listed in Certificate of Compliance				0.372 (0.945) (min.)	0.3310 (0.8407) (max.)		0.3232 (0.8209) (max.)	150 (381) (max.)	0.014 (0.0356) (min.)
bounding dimensions (17x17B06)	0.9311†	0.9268	0.0008	0.372 (0.945)	0.3310 (0.8407)	0.0205 (0.0521)	0.3232 (0.8209)	150 (381)	0.014 (0.0356)

† The k_{eff} value listed for the 17x17B04 case is higher than that for the case with the bounding dimensions. Therefore, the 0.9324 value from case 17x17B04 is listed in Table 6.1.1 as the maximum.

Table 6.2.19
MAXIMUM K_{EFF} VALUES FOR THE 17X17C ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

17x17C (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.02 g/cm^2) 264 fuel rods, 25 guide tubes, pitch=0.502 (1.275), Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17C01	0.9293	0.9250	0.0008	0.379 (0.963)	0.3310 (0.8407)	0.0240 (0.0610)	0.3232 (0.8209)	150 (381)	0.020 (0.0508)
17x17C02	0.9336	0.9293	0.0008	0.377 (0.958)	0.3330 (0.8458)	0.0220 (0.0559)	0.3252 (0.8260)	150 (381)	0.020 (0.0508)
Dimensions Listed in Certificate of Compliance				0.377 (0.958) (min.)	0.3330 (0.8458) (max.)		0.3252 (0.8260) (max.)	150 (381) (max.)	0.020 (0.0508) (min.)
bounding dimensions (17x17C02)	0.9336	0.9293	0.0008	0.377 (0.958)	0.3330 (0.8458)	0.0220 (0.0559)	0.3252 (0.8260)	150 (381)	0.020 (0.0508)

Table 6.2.20
MAXIMUM K_{EFF} VALUES FOR THE 7X7B ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

7x7B (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 49 fuel rods, 0 water rods, pitch=0.738 (1.875), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
7x7B01	0.9372	0.9330	0.0007	0.5630 (1.4300)	0.4990 (1.2675)	0.0320 (0.0813)	0.4870 (1.2370)	150 (381)	n/a	0.080 (0.203)
7x7B02	0.9301	0.9260	0.0007	0.5630 (1.4300)	0.4890 (1.2421)	0.0370 (0.0940)	0.4770 (1.2116)	150 (381)	n/a	0.102 (0.259)
7x7B03	0.9313	0.9271	0.0008	0.5630 (1.4300)	0.4890 (1.2421)	0.0370 (0.0940)	0.4770 (1.2116)	150 (381)	n/a	0.080 (0.203)
7x7B04	0.9311	0.9270	0.0007	0.5700 (1.4478)	0.4990 (1.2675)	0.0355 (0.0902)	0.4880 (1.2395)	150 (381)	n/a	0.080 (0.203)
7x7B05	0.9350	0.9306	0.0008	0.5630 (1.4300)	0.4950 (1.2573)	0.0340 (0.0864)	0.4775 (1.2129)	150 (381)	n/a	0.080 (0.203)
7x7B06	0.9298	0.9260	0.0006	0.5700 (1.4478)	0.4990 (1.2675)	0.0355 (0.0902)	0.4910 (1.2471)	150 (381)	n/a	0.080 (0.203)
Dimensions Listed in Certificate of Compliance				0.5630 (1.4300) (min.)	0.4990 (1.2675) (max.)		0.4910 (1.2471) (max.)	150 (381) (max.)	n/a	0.120 (0.305) (max.)
bounding dimensions (B7x7B01)	0.9375	0.9332	0.0008	0.5630 (1.4300)	0.4990 (1.2675)	0.0320 (0.0813)	0.4910 (1.2471)	150 (381)	n/a	0.102 (0.259)
bounding dimensions with 120 mil channel (B7x7B02)	0.9386	0.9344	0.0007	0.5630 (1.4300)	0.4990 (1.2675)	0.0320 (0.0813)	0.4910 (1.2471)	150 (381)	n/a	0.120 (0.305)

Table 6.2.21
MAXIMUM K_{EFF} VALUES FOR THE 8X8B ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

8x8B (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 63 or 64 fuel rods†, 1 or 0 water rods†, pitch† = 0.636-0.642 (1.615-1.631), Zr clad												
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	Fuel rods	pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8B01	0.9310	0.9265	0.0009	63	0.641 (1.628)	0.4840 (1.2294)	0.4140 (1.0516)	0.0350 (0.0889)	0.4050 (1.0287)	150 (381)	0.035 (0.0889)	0.100 (0.254)
8x8B02	0.9227	0.9185	0.0007	63	0.636 (1.615)	0.4840 (1.2294)	0.4140 (1.0516)	0.0350 (0.0889)	0.4050 (1.0287)	150 (381)	0.035 (0.0889)	0.100 (0.254)
8x8B03	0.9299	0.9257	0.0008	63	0.640 (1.626)	0.4930 (1.2522)	0.4250 (1.0795)	0.0340 (0.0864)	0.4160 (1.0566)	150 (381)	0.034 (0.0864)	0.100 (0.254)
8x8B04	0.9236	0.9194	0.0008	64	0.642 (1.631)	0.5015 (1.2738)	0.4295 (1.0909)	0.0360 (0.0914)	0.4195 (1.0655)	150 (381)	n/a	0.100 (0.254)
Dimensions Listed in Certificate of Compliance				63 or 64	0.636-0.642 (1.615-1.631)	0.4840 (1.2294) (min.)	0.4295 (1.0909) (max.)		0.4195 (1.0655) (max.)	150 (381) (max.)	0.034 (0.0864)	0.120 (0.305) (max.)
bounding (pitch=0.636) (B8x8B01)	0.9346	0.9301	0.0009	63	0.636 (1.615)	0.4840 (1.2294)	0.4295 (1.0909)	0.02725 (0.0692)	0.4195 (1.0655)	150 (381)	0.034 (0.0864)	0.120 (0.305)
bounding (pitch=0.640) (B8x8B02)	0.9385	0.9343	0.0008	63	0.640 (1.626)	0.4840 (1.2294)	0.4295 (1.0909)	0.02725 (0.0692)	0.4195 (1.0655)	150 (381)	0.034 (0.0864)	0.120 (0.305)
bounding (pitch=0.642) (B8x8B03)	0.9416	0.9375	0.0007	63	0.642 (1.631)	0.4840 (1.2294)	0.4295 (1.0909)	0.02725 (0.0692)	0.4195 (1.0655)	150 (381)	0.034 (0.0864)	0.120 (0.305)

† This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

Table 6.2.22
MAXIMUM K_{EFF} VALUES FOR THE 8X8C ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

8x8C (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 62 fuel rods, 2 water rods, pitch† = 0.636-0.641(1.615-1.628) , Zr clad											
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8C01	0.9315	0.9273	0.0007	0.641 (1.628)	0.4840 (1.2294)	0.4140 (1.0516)	0.0350 (0.0889)	0.4050 (1.0287)	150 (381)	0.035 (0.0889)	0.100 (0.254)
8x8C02	0.9313	0.9268	0.0009	0.640 (1.626)	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4100 (1.0414)	150 (381)	0.030 (0.0762)	0.000 (0.000)
8x8C03	0.9329	0.9286	0.0008	0.640 (1.626)	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4100 (1.0414)	150 (381)	0.030 (0.0762)	0.080 (0.203)
8x8C04	0.9348††	0.9307	0.0007	0.640 (1.626)	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4100 (1.0414)	150 (381)	0.030 (0.0762)	0.100 (0.254)
8x8C05	0.9353	0.9312	0.0007	0.640 (1.626)	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4100 (1.0414)	150 (381)	0.030 (0.0762)	0.120 (0.305)
8x8C06	0.9353	0.9312	0.0007	0.640 (1.626)	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	0.030 (0.0762)	0.100 (0.254)
8x8C07	0.9314	0.9273	0.0007	0.640 (1.626)	0.4830 (1.2268)	0.4150 (1.0541)	0.0340 (0.0864)	0.4100 (1.0414)	150 (381)	0.030 (0.0762)	0.100 (0.254)
8x8C08	0.9339	0.9298	0.0007	0.640 (1.626)	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4100 (1.0414)	150 (381)	0.034 (0.0864)	0.100 (0.254)
8x8C09	0.9301	0.9260	0.0007	0.640 (1.626)	0.4930 (1.2522)	0.4250 (1.0795)	0.0340 (0.0864)	0.4160 (1.0566)	150 (381)	0.034 (0.0864)	0.100 (0.254)
8x8C10	0.9317	0.9275	0.0008	0.640 (1.626)	0.4830 (1.2268)	0.4150 (1.0541)	0.0340 (0.0864)	0.4100 (1.0414)	150 (381)	0.030 (0.0762)	0.120 (0.305)
8x8C11	0.9328	0.9287	0.0007	0.640 (1.626)	0.4830 (1.2268)	0.4150 (1.0541)	0.0340 (0.0864)	0.4100 (1.0414)	150 (381)	0.030 (0.0762)	0.120 (0.305)
8x8C12	0.9285	0.9242	0.0008	0.636 (1.615)	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	0.030 (0.0762)	0.120 (0.305)
Dimensions Listed in Certificate of Compliance				0.636-0.641 (1.615-1.628)	0.4830 (1.2268) (min.)	0.4250 (1.0795) (max.)		0.4160 (1.0566) (max.)	150 (381) (max.)	0.000 (0.000) (min.)	0.120 (0.305) (max.)
bounding (pitch=0.636) (B8x8C01)	0.9357	0.9313	0.0009	0.636 (1.615)	0.4830 (1.2268)	0.4250 (1.0795)	0.0290 (0.0737)	0.4160 (1.0566)	150 (381)	0.000 (0.000)	0.120 (0.305)
bounding (pitch=0.640) (B8x8C02)	0.9425	0.9384	0.0007	0.640 (1.626)	0.4830 (1.2268)	0.4250 (1.0795)	0.0290 (0.0737)	0.4160 (1.0566)	150 (381)	0.000 (0.000)	0.120 (0.305)
Bounding (pitch=0.641) (B8x8C03)	0.9418	0.9375	0.0008	0.641 (1.628)	0.4830 (1.2268)	0.4250 (1.0795)	0.0290 (0.0737)	0.4160 (1.0566)	150 (381)	0.000 (0.000)	0.120 (0.305)

Table 6.2.23

† This assembly class was analyzed and qualified for a small variation in the pitch.
†† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9343.

MAXIMUM K_{EFF} VALUES FOR THE 8X8D ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

8x8D (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 60-61 fuel rods, 1-4 water rods†, pitch=0.640 (1.626), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8D01	0.9342	0.9302	0.0006	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	0.03/0.025 (0.076/0.064)	0.100 (0.254)
8x8D02	0.9325	0.9284	0.0007	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	0.030 (0.076)	0.100 (0.254)
8x8D03	0.9351	0.9309	0.0008	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	0.025 (0.064)	0.100 (0.254)
8x8D04	0.9338	0.9296	0.0007	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	0.040 (0.102)	0.100 (0.254)
8x8D05	0.9339	0.9294	0.0009	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4100 (1.0414)	150 (381)	0.040 (0.102)	0.100 (0.254)
8x8D06	0.9365	0.9324	0.0007	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	0.040 (0.102)	0.120 (0.305)
8x8D07	0.9341	0.9297	0.0009	0.4830 (1.2268)	0.4190 (1.0643)	0.0320 (0.0813)	0.4110 (1.0439)	150 (381)	0.040 (0.102)	0.080 (0.203)
8x8D08	0.9376	0.9332	0.0009	0.4830 (1.2268)	0.4230 (1.0744)	0.0300 (0.0762)	0.4140 (1.0516)	150 (381)	0.030 (0.076)	0.080 (0.203)
Dimensions Listed in Certificate of Compliance				0.4830 (1.2268) (min.)	0.4230 (1.0744) (max.)		0.4140 (1.0516) (max.)	150 (381) (max.)	0.000 (0.000) (min.)	0.120 (0.305) (max.)
bounding dimensions (B8x8D01)	0.9403	0.9363	0.0007	0.4830 (1.2268)	0.4230 (1.0744)	0.0300 (0.0762)	0.4140 (1.0516)	150 (381)	0.000 (0.000)	0.120 (0.305)

† Fuel assemblies 8x8D01 through 8x8D03 have 4 water rods that are similar in size to the fuel rods, while assemblies 8x8D04 through 8x8D07 have 1 large water rod that takes the place of the 4 water rods. Fuel assembly 8x8D08 contains 3 water rods that are similar in size to the fuel rods.

Table 6.2.24
MAXIMUM K_{EFF} VALUES FOR THE 8X8E ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

8x8E (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 59 fuel rods, 5 water rods, pitch=0.640 (1.626), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8E01	0.9312	0.9270	0.0008	0.4930 (1.2522)	0.4250 (1.0795)	0.0340 (0.0864)	0.4160 (1.0566)	150 (381)	0.034 (0.086)	0.100 (0.254)
Dimensions Listed in Certificate of Compliance				0.4930 (1.2522) (min.)	0.4250 (1.0795) (max.)		0.4160 (1.0566) (max.)	150 (381) (max.)	0.034 (0.086) (min.)	0.100 (0.254) (max.)

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Table 6.2.25
MAXIMUM K_{EFF} VALUES FOR THE 8X8F ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

8x8F (3.6% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 64 fuel rods, 4 rectangular water cross segments dividing the assembly into four quadrants, pitch=0.609 (1.547), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8F01	0.9153	0.9111	0.0007	0.4576 (1.1623)	0.3996 (1.0150)	0.0290 (0.0737)	0.3913 (0.9939)	150 (381)	0.0315 (0.0800)	0.055 (0.140)
Dimensions Listed in Certificate of Compliance				0.4576 (1.1623) (min.)	0.3996 (1.0150) (max.)		0.3913 (0.9939) (max.)	150 (381) (max.)	0.0315 (0.0800) (min.)	0.055 (0.140) (max.)

Table 6.2.26
MAXIMUM K_{EFF} VALUES FOR THE 9X9A ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

9x9A (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 74/66 fuel rods†, 2 water rods, pitch=0.566 (1.438), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9A01 (axial segment with all rods)	0.9353	0.9310	0.0008	0.4400 (1.1176)	0.3840 (0.9754)	0.0280 (0.0711)	0.3760 (0.9550)	150 (381)	0.030 (0.076)	0.100 (0.254)
9x9A02 (axial segment with only the full length rods)	0.9388	0.9345	0.0008	0.4400 (1.1176)	0.3840 (0.9754)	0.0280 (0.0711)	0.3760 (0.9550)	150 (381)	0.030 (0.076)	0.100 (0.254)
9x9A03 (actual three-dimensional representation of all rods)	0.9351	0.9310	0.0007	0.4400 (1.1176)	0.3840 (0.9754)	0.0280 (0.0711)	0.3760 (0.9550)	150/90 (381/228.6)	0.030 (0.076)	0.100 (0.254)
9x9A04 (axial segment with only the full length rods)	0.9396	0.9355	0.0007	0.4400 (1.1176)	0.3840 (0.9754)	0.0280 (0.0711)	0.3760 (0.9550)	150 (381)	0.030 (0.076)	0.120 (0.305)
Dimensions Listed in Certificate of Compliance				0.4400 (1.1176) (min.)	0.3840 (0.9754) (max.)		0.3760 (0.9550) (max.)	150 (381) (max.)	0.000 (0.000) (min.)	0.120 (0.305) (max.)
bounding dimensions (axial segment with only the full length rods) (B9x9A01)	0.9417	0.9374	0.0008	0.4400 (1.1176)	0.3840 (0.9754)	0.0280 (0.0711)	0.3760 (0.9550)	150 (381)	0.000 (0.000)	0.120 (0.305)

† This assembly class contains 66 full length rods and 8 partial length rods. In order to eliminate a requirement on the length of the partial length rods, separate calculations were performed for the axial segments with and without the partial length rods.

Table 6.2.27
MAXIMUM K_{EFF} VALUES FOR THE 9X9B ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

9x9B (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 72 fuel rods, 1 water rod (square, replacing 9 fuel rods), pitch=0.569 (1.445) to 0.572 (1.453)†, Zr clad											
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9B01	0.9380	0.9336	0.0008	0.569 (1.445)	0.4330 (1.0998)	0.3807 (0.9670)	0.0262 (0.0665)	0.3737 (0.9492)	150 (381)	0.0285 (0.0724)	0.100 (0.254)
9x9B02	0.9373	0.9329	0.0009	0.569 (1.445)	0.4330 (1.0998)	0.3810 (0.9677)	0.0260 (0.0660)	0.3737 (0.9492)	150 (381)	0.0285 (0.0724)	0.100 (0.254)
9x9B03	0.9417	0.9374	0.0008	0.572 (1.453)	0.4330 (1.0998)	0.3810 (0.9677)	0.0260 (0.0660)	0.3737 (0.9492)	150 (381)	0.0285 (0.0724)	0.100 (0.254)
Dimensions Listed in Certificate of Compliance				0.572 (1.453)	0.4330 (1.0998) (min.)	0.3810 (0.9677) (max.)		0.3740 (0.9500) (max.)	150 (381) (max.)	0.000 (0.000) (min.)	0.120 (0.305) (max.)
bounding dimensions (B9x9B01)	0.9436	0.9394	0.0008	0.572 (1.453)	0.4330 (1.0998)	0.3810 (0.9677)	0.0260 (0.0660)	0.3740 (0.9500) ††	150 (381)	0.000 (0.000)	0.120 (0.305)

† This assembly class was analyzed and qualified for a small variation in the pitch.
†† This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.28
MAXIMUM K_{EFF} VALUES FOR THE 9X9C ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

9x9C (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 80 fuel rods, 1 water rods, pitch=0.572(1.453), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9C01	0.9395	0.9352	0.0008	0.4230 (1.0744)	0.3640 (0.9246)	0.0295 (0.0749)	0.3565 (0.9055)	150 (381)	0.020 (0.051)	0.100 (0.254)
Dimensions Listed in Certificate of Compliance				0.4230 (1.0744) (min.)	0.3640 (0.9246) (max.)		0.3565 (0.9055) (max.)	150 (381) (max.)	0.020 (0.051) (min.)	0.100 (0.254) (max.)

Table 6.2.29
MAXIMUM K_{EFF} VALUES FOR THE 9X9D ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

9x9D (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 79 fuel rods, 2 water rods, pitch=0.572 (1.453), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9D01	0.9394	0.9350	0.0009	0.4240 (1.0770)	0.3640 (0.9246)	0.0300 (0.0762)	0.3565 (0.9055)	150 (381)	0.0300 (0.0762)	0.100 (0.254)
Dimensions Listed in Certificate of Compliance				0.4240 (1.0770) (min.)	0.3640 (0.9246) (max.)		0.3565 (0.9055) (max.)	150 (381) (max.)	0.0300 (0.0762) (min.)	0.100 (0.254) (max.)

Table 6.2.30
MAXIMUM K_{EFF} VALUES FOR THE 9X9E ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

9x9E (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 76 fuel rods, 5 water rods, pitch=0.572 (1.453), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9E01	0.9402	0.9359	0.0008	0.4170 (1.0592)	0.3640 (0.9246)	0.0265 (0.0673)	0.3530 (0.8966)	150 (381)	0.0120 (0.0305)	0.120 (0.305)
9x9E02	0.9424	0.9380	0.0008	0.4170 (1.0592) 0.4430 (1.1252)	0.3640 (0.9246) 0.3860 (0.9804)	0.0265 (0.0673) 0.0285 (0.0724)	0.3530 (0.8966) 0.3745 (0.9512)	150 (381)	0.0120 (0.0305)	0.120 (0.305)
Dimensions Listed in Certificate of Compliance†				0.4170 (1.0592) (min.)	0.3640 (0.9246) (max.)		0.3530 (0.8966) (max.)	150 (381) (max.)	0.0120 (0.0305) (min.)	0.120 (0.305) (max.)
bounding dimensions (9x9E02)	0.9424	0.9380	0.0008	0.4170 (1.0592) 0.4430 (1.1252)	0.3640 (0.9246) 0.3860 (0.9804)	0.0265 (0.0673) 0.0285 (0.0724)	0.3530 (0.8966) 0.3745 (0.9512)	150 (381)	0.0120 (0.0305)	0.120 (0.305)

† This fuel assembly, also known as SPC 9x9-5, contains fuel rods with different cladding and pellet diameters which do not bound each other. To be consistent in the way fuel assemblies are listed in the Certificate of Compliance, two assembly classes (9x9E and 9x9F) are required to specify this assembly. Each class contains the actual geometry (9x9E02 and 9x9F02), as well as a hypothetical geometry with either all small rods (9x9E01) or all large rods (9x9F01). The Certificate of Compliance lists the small rod dimensions for class 9x9E and the large rod dimensions for class 9x9F, and a note that both classes are used to qualify the assembly. The analyses demonstrate that all configurations, including the actual geometry, are acceptable.

Table 6.2.31
MAXIMUM K_{EFF} VALUES FOR THE 9X9F ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

9x9F (4.1% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 76 fuel rods, 5 water rods, pitch=0.572 (1.453), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9F01	0.9369	0.9326	0.0007	0.4430 (1.1252)	0.3860 (0.9804)	0.0285 (0.0724)	0.3745 (0.9512)	150 (381)	0.0120 (0.0305)	0.120 (0.305)
9x9F02	0.9424	0.9380	0.0008	0.4170 (1.0592) 0.4430 (1.1252)	0.3640 (0.9246) 0.3860 (0.9804)	0.0265 (0.0673) 0.0285 (0.0724)	0.3530 (0.8966) 0.3745 (0.9512)	150 (381)	0.0120 (0.0305)	0.120 (0.305)
Dimensions Listed in Certificate of Compliance†				0.4430 (1.1252) (min.)	0.3860 (0.9804) (max.)		0.3745 (0.9512) (max.)	150 (381) (max.)	0.0120 (0.0305) (min.)	0.120 (0.305) (max.)
bounding dimensions (9x9F02)	0.9424	0.9380	0.0008	0.4170 (1.0592) 0.4430 (1.1252)	0.3640 (0.9246) 0.3860 (0.9804)	0.0265 (0.0673) 0.0285 (0.0724)	0.3530 (0.8966) 0.3745 (0.9512)	150 (381)	0.0120 (0.0305)	0.120 (0.305)

† This fuel assembly, also known as SPC 9x9-5, contains fuel rods with different cladding and pellet diameters which do not bound each other. To be consistent in the way fuel assemblies are listed in the Certificate of Compliance, two assembly classes (9x9E and 9x9F) are required to specify this assembly. Each class contains the actual geometry (9x9E02 and 9x9F02), as well as a hypothetical geometry with either all small rods (9x9E01) or all large rods (9x9F01). The Certificate of Compliance lists the small rod dimensions for class 9x9E and the large rod dimensions for class 9x9F, and a note that both classes are used to qualify the assembly. The analyses demonstrate that all configurations, including the actual geometry, are acceptable.

Table 6.2.32
MAXIMUM K_{EFF} VALUES FOR THE 10X10A ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

10x10A (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 92/78 fuel rods†, 2 water rods, pitch=0.510(1.295), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10A01 (axial segment with all rods)	0.9377	0.9335	0.0008	0.4040 (1.0262)	0.3520 (0.8941)	0.0260 (0.0660)	0.3450 (0.8763)	155 (393.7)	0.030 (0.076)	0.100 (0.254)
10x10A02 (axial segment with only the full length rods)	0.9426	0.9386	0.0007	0.4040 (1.0262)	0.3520 (0.8941)	0.0260 (0.0660)	0.3450 (0.8763)	155 (393.7)	0.030 (0.076)	0.100 (0.254)
10x10A03 (actual three-dimensional representation of all rods)	0.9396	0.9356	0.0007	0.4040 (1.0262)	0.3520 (0.8941)	0.0260 (0.0660)	0.3450 (0.8763)	155/90 (393.7/228.6)	0.030 (0.076)	0.100 (0.254)
Dimensions Listed in Certificate of Compliance				0.4040 (1.0262) (min.)	0.3520 (0.8941) (max.)		0.3455 (0.8776) (max.)	150 (381)†† (max.)	0.030 (0.076) (min.)	0.120 (0.305) (max.)
bounding dimensions (axial segment with only the full length rods) (B10x10A01)	0.9457†††	0.9414	0.0008	0.4040 (1.0262)	0.3520 (0.8941)	0.0260 (0.0660)	0.3455 (0.8776) ‡	155 (393.7)	0.030 (0.076)	0.120 (0.305)

† This assembly class contains 78 full-length rods and 14 partial-length rods. In order to eliminate the requirement on the length of the partial length rods, separate calculations were performed for axial segments with and without the partial length rods.

†† Although the analysis qualifies this assembly for a maximum active fuel length of 155 inches (393.7 cm), the Certificate of Compliance limits the active fuel length to 150 inches (381 cm). This is due to the fact that the fixed neutron absorber panels are 156 inches (396.24 cm) in length.

††† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9453.

‡ This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.33
MAXIMUM K_{EFF} VALUES FOR THE 10X10B ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

10x10B (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 91/83 fuel rods†, 1 water rods (square, replacing 9 fuel rods), pitch=0.510 (1.295), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10B01 (axial segment with all rods)	0.9384	0.9341	0.0008	0.3957 (1.0051)	0.3480 (0.8839)	0.0239 (0.0607)	0.3413 (0.8669)	155 (393.7)	0.0285 (0.0724)	0.100 (0.254)
10x10B02 (axial segment with only the full length rods)	0.9416	0.9373	0.0008	0.3957 (1.0051)	0.3480 (0.8839)	0.0239 (0.0607)	0.3413 (0.8669)	155 (393.7)	0.0285 (0.0724)	0.100 (0.254)
10x10B03 (actual three-dimensional representation of all rods)	0.9375	0.9334	0.0007	0.3957 (1.0051)	0.3480 (0.8839)	0.0239 (0.0607)	0.3413 (0.8669)	155/90 (393.7/228.6)	0.0285 (0.0724)	0.100 (0.254)
Dimensions Listed in Certificate of Compliance				0.3957 (1.0051) (min.)	0.3480 (0.8839) (max.)		0.3420 (0.8687) (max.)	150 (381)†† (max.)	0.000 (0.000) (min.)	0.120 (0.305) (max.)
bounding dimensions (axial segment with only the full length rods) (B10x10B01)	0.9436	0.9395	0.0007	0.3957 (1.0051)	0.3480 (0.8839)	0.0239 (0.0607)	0.3420 (0.8687) †††	155 (393.7)	0.000 (0.000)	0.120 (0.305)

† This assembly class contains 83 full length rods and 8 partial length rods. In order to eliminate a requirement on the length of the partial length rods, separate calculations were performed for the axial segments with and without the partial length rods.

†† Although the analysis qualifies this assembly for a maximum active fuel length of 155 (393.7 cm) inches, the Certificate of Compliance limits the active fuel length to 150 (381 cm) inches. This is due to the fact that the fixed neutron absorber panels are 156 inches (396.24 cm) in length.

††† This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.34
MAXIMUM K_{EFF} VALUES FOR THE 10X10C ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

10x10C (4.2% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 96 fuel rods, 5 water rods (1 center diamond and 4 rectangular), pitch=0.488 (1.240), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10C01	0.9433	0.9392	0.0007	0.3780 (0.9601)	0.3294 (0.8367)	0.0243 (0.0617)	0.3224 (0.8189)	150 (381)	0.031 (0.079)	0.055 (0.140)
Dimensions Listed in Certificate of Compliance				0.3780 (0.9601) (min.)	0.3294 (0.8367) (max.)		0.3224 (0.8189) (max.)	150 (381) (max.)	0.031 (0.079) (min.)	0.055 (0.140) (max.)

Table 6.2.35
MAXIMUM K_{EFF} VALUES FOR THE 10X10D ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

10x10D (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 100 fuel rods, 0 water rods, pitch=0.565 (1.435), SS clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10D01	0.9376	0.9333	0.0008	0.3960 (1.0058)	0.3560 (0.9042)	0.0200 (0.0508)	0.350 (0.889)	83 (210.8)	n/a	0.080 (0.203)
Dimensions Listed in Certificate of Compliance				0.3960 (1.0058) (min.)	0.3560 (0.9042) (max.)		0.350 (0.889) (max.)	83 (210.8) (max.)	n/a	0.080 (0.203) (max.)

Table 6.2.36
MAXIMUM K_{EFF} VALUES FOR THE 10X10E ASSEMBLY CLASS IN THE MPC-68
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

10x10E (4.0% Enrichment, fixed neutron absorber ^{10}B minimum loading of 0.0279 g/cm^2) 96 fuel rods, 4 water rods, pitch=0.557 (1.415), SS clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10E01	0.9185	0.9144	0.0007	0.3940 (1.0008)	0.3500 (0.8890)	0.0220 (0.0559)	0.3430 (0.8712)	83 (210.8)	0.022 (0.056)	0.080 (0.203)
Dimensions Listed in Certificate of Compliance				0.3940 (1.0008) (min.)	0.3500 (0.8890) (max.)		0.3430 (0.8712) (max.)	83 (210.8) (max.)	0.022 (0.056) (min.)	0.080 (0.203) (max.)

Table 6.2.37
MAXIMUM K_{EFF} VALUES FOR THE 6X6A ASSEMBLY CLASS IN THE MPC-68F
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

6x6A (3.0% Enrichment†, fixed neutron absorber ^{10}B minimum loading of 0.0067 g/cm^2) 35 or 36 fuel rods††, 1 or 0 water rods††, pitch††=0.694 (1.763) to 0.710 (1.803), Zr clad												
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	pitch	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6A01	0.7539	0.7498	0.0007	0.694 (1.763)	36	0.5645 (1.4338)	0.4945 (1.2560)	0.0350 (0.0889)	0.4940 (1.2548)	110 (279.4)	n/a	0.060 (0.152)
6x6A02	0.7517	0.7476	0.0007	0.694 (1.763)	36	0.5645 (1.4338)	0.4925 (1.2510)	0.0360 (0.0914)	0.4820 (1.2243)	110 (279.4)	n/a	0.060 (0.152)
6x6A03	0.7545	0.7501	0.0008	0.694 (1.763)	36	0.5645 (1.4338)	0.4945 (1.2560)	0.0350 (0.0889)	0.4820 (1.2243)	110 (279.4)	n/a	0.060 (0.152)
6x6A04	0.7537	0.7494	0.0008	0.694 (1.763)	36	0.5550 (1.4097)	0.4850 (1.2319)	0.0350 (0.0889)	0.4820 (1.2243)	110 (279.4)	n/a	0.060 (0.152)
6x6A05	0.7555	0.7512	0.0008	0.696 (1.768)	36	0.5625 (1.4288)	0.4925 (1.2510)	0.0350 (0.0889)	0.4820 (1.2243)	110 (279.4)	n/a	0.060 (0.152)
6x6A06	0.7618	0.7576	0.0008	0.696 (1.768)	35	0.5625 (1.4288)	0.4925 (1.2510)	0.0350 (0.0889)	0.4820 (1.2243)	110 (279.4)	0.0 (0.0)	0.060 (0.152)
6x6A07	0.7588	0.7550	0.0007	0.700 (1.778)	36	0.5555 (1.4110)	0.4850 (1.2319)	0.03525 (0.0895)	0.4780 (1.2141)	110 (279.4)	n/a	0.060 (0.152)
6x6A08	0.7808	0.7766	0.0007	0.710 (1.803)	36	0.5625 (1.4288)	0.5105 (1.2967)	0.0260 (0.0660)	0.4980 (1.2649)	110 (279.4)	n/a	0.060 (0.152)
Dimensions Listed in Certificate of Compliance				0.710 (1.803) (max.)	35 or 36	0.5550 (1.4097) (min.)	0.5105 (1.2967) (max.)	0.02225 (0.0565)	0.4980 (1.2649) (max.)	120 (304.8) (max.)	0.0 (0.0)	0.060 (0.152) (max.)
bounding dimensions (B6x6A01)	0.7727	0.7685	0.0007	0.694 (1.763)	35	0.5550 (1.4097)	0.5105 (1.2967)	0.02225 (0.0565)	0.4980 (1.2649)	120 (304.8)	0.0 (0.0)	0.060 (0.152)
bounding dimensions (B6x6A02)	0.7782	0.7738	0.0008	0.700 (1.778)	35	0.5550 (1.4097)	0.5105 (1.2967)	0.02225 (0.0565)	0.4980 (1.2649)	120 (304.8)	0.0 (0.0)	0.060 (0.152)
bounding dimensions (B6x6A03)	0.7888	0.7846	0.0007	0.710 (1.803)	35	0.5550 (1.4097)	0.5105 (1.2967)	0.02225 (0.0565)	0.4980 (1.2649)	120 (304.8)	0.0 (0.0)	0.060 (0.152)

† Although the calculations were performed for 3.0%, the enrichment is limited in the Certificate of Compliance to 2.7%.
†† This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

Table 6.2.38
MAXIMUM K_{EFF} VALUES FOR THE 6X6B ASSEMBLY CLASS IN THE MPC-68F
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

6x6B (3.0% Enrichment†, fixed neutron absorber ^{10}B minimum loading of 0.0067 g/cm^2) 35 or 36 fuel rods†† (up to 9 MOX rods), 1 or 0 water rods††, pitch††=0.694(1.763) to 0.710 (1.803), Zr clad												
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	pitch	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6B01	0.7604	0.7563	0.0007	0.694 (1.763)	36	0.5645 (1.4338)	0.4945 (1.2560)	0.0350 (0.0889)	0.4820 (1.2243)	110 (279.4)	n/a	0.060 (0.152)
6x6B02	0.7618	0.7577	0.0007	0.694 (1.763)	36	0.5625 (1.4288)	0.4925 (1.2510)	0.0350 (0.0889)	0.4820 (1.2243)	110 (279.4)	n/a	0.060 (0.152)
6x6B03	0.7619	0.7578	0.0007	0.696 (1.768)	36	0.5625 (1.4288)	0.4925 (1.2510)	0.0350 (0.0889)	0.4820 (1.2243)	110 (279.4)	n/a	0.060 (0.152)
6x6B04	0.7686	0.7644	0.0008	0.696 (1.768)	35	0.5625 (1.4288)	0.4925 (1.2510)	0.0350 (0.0889)	0.4820 (1.2243)	110 (279.4)	0.0 (0.0)	0.060 (0.152)
6x6B05	0.7824	0.7785	0.0006	0.710 (1.803)	35	0.5625 (1.4288)	0.4925 (1.2510)	0.0350 (0.0889)	0.4820 (1.2243)	110 (279.4)	0.0 (0.0)	0.060 (0.152)
Dimensions Listed in Certificate of Compliance				0.710 (1.803) (max.)	35 or 36	0.5625 (1.4288) (min.)	0.4945 (1.2560) (max.)		0.4820 (1.2243) (max.)	120 (304.8) (max.)	0.0 (0.0)	0.060 (0.152) (max.)
bounding dimensions (B6x6B01)	0.7822†††	0.7783	0.0006	0.710 (1.803)	35	0.5625 (1.4288)	0.4945 (1.2560)	0.0340 (0.0864)	0.4820 (1.2243)	120 (304.8)	0.0 (0.0)	0.060 (0.152)

Note:

1. These assemblies contain up to 9 MOX pins. The composition of the MOX fuel pins is given in Table 6.3.4.

† The ^{235}U enrichment of the MOX and UO_2 pins is assumed to be 0.711% and 3.0%, respectively.

†† This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

††† The k_{eff} value listed for the 6x6B05 case is slightly higher than that for the case with the bounding dimensions. However, the difference (0.0002) is well within the statistical uncertainties, and thus, the two values are statistically equivalent (within 1σ). Therefore, the 0.7824 value is listed in Tables 6.1.2 and 6.1.3 as the maximum.

Table 6.2.39
MAXIMUM K_{EFF} VALUES FOR THE 6X6C ASSEMBLY CLASS IN THE MPC-68F
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

6x6C (3.0% Enrichment†, fixed neutron absorber ^{10}B minimum loading of 0.0067 g/cm^2) 36 fuel rods, 0 water rods, pitch=0.740 (1.880), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6C01	0.8021	0.7980	0.0007	0.5630 (1.4300)	0.4990 (1.2675)	0.0320 (0.0813)	0.4880 (1.2395)	77.5 (196.85)	n/a	0.060 (0.152)
Dimensions Listed in Certificate of Compliance				0.5630 (1.4300) (min.)	0.4990 (1.2675) (max.)		0.4880 (1.2395) (max.)	77.5 (196.85) (max.)	n/a	0.060 (0.152) (max.)

† Although the calculations were performed for 3.0%, the enrichment is limited in the Certificate of Compliance to 2.7%.

Table 6.2.40
MAXIMUM K_{EFF} VALUES FOR THE 7X7A ASSEMBLY CLASS IN THE MPC-68F
(all dimensions are in inches)

7x7A (3.0% Enrichment†, fixed neutron absorber ^{10}B minimum loading of 0.0067 g/cm^2) 49 fuel rods, 0 water rods, pitch=0.631 (1.603), Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
7x7A01	0.7974	0.7932	0.0008	0.4860 (1.2344)	0.4204 (1.0678)	0.0328 (0.0833)	0.4110 (1.0439)	80 (203.2)	n/a	0.060 (0.152)
Dimensions Listed in Certificate of Compliance				0.4860 (1.2344) (min.)	0.4204 (1.0678) (max.)		0.4110 (1.0439) (max.)	80 (203.2) (max.)	n/a	0.060 (0.152) (max.)

† Although the calculations were performed for 3.0%, the enrichment is limited in the Certificate of Compliance to 2.7%.

Table 6.2.41
MAXIMUM K_{EFF} VALUES FOR THE 8X8A ASSEMBLY CLASS IN THE MPC-68F
(all dimensions are in inches and centimeters, the values in parentheses are in centimeters)

8x8A (3.0% Enrichment†, fixed neutron absorber ^{10}B minimum loading of 0.0067 g/cm^2) 63 or 64 fuel rods††, 0 water rods, pitch=0.523 (1.328), Zr clad											
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8A01	0.7685	0.7644	0.0007	64	0.4120 (1.0465)	0.3620 (0.9195)	0.0250 (0.0635)	0.3580 (0.9093)	110 (279.4)	n/a	0.100 (0.254)
8x8A02	0.7697	0.7656	0.0007	63	0.4120 (1.0465)	0.3620 (0.9195)	0.0250 (0.0635)	0.3580 (0.9093)	120 (304.8)	n/a	0.100 (0.254)
Dimensions Listed in Certificate of Compliance				63	0.4120 (1.0465) (min.)	0.3620 (0.9195) (max.)		0.3580 (0.9093) (max.)	120 (304.8) (max.)	n/a	0.100 (0.254) (max.)
bounding dimensions (8x8A02)	0.7697	0.7656	0.0007	63	0.4120 (1.0465)	0.3620 (0.9195)	0.0250 (0.0635)	0.3580 (0.9093)	120 (304.8)	n/a	0.100 (0.254)

† Although the calculations were performed for 3.0%, the enrichment is limited in the Certificate of Compliance to 2.7%.
†† This assembly class was analyzed and qualified for a variation in the number of fuel rods.

Table 6.2.42

SPECIFICATION OF THE THORIA ROD CANISTER AND THE THORIA RODS

Canister ID	4.81” (12.22 cm)
Canister Wall Thickness	0.11” (0.28 cm)
Separator Assembly Plates Thickness	0.11” (0.28 cm)
Cladding OD	0.412” (1.046 cm)
Cladding ID	0.362” (0.919 cm)
Pellet OD	0.358” (0.909 cm)
Active Length	110.5” (280.67 cm)
Fuel Composition	1.5% UO_2 and 98.5% ThO_2
Initial Enrichment	93.5 wt% ^{235}U for 1.5% of the fuel
Maximum k_{eff} †	0.1813
Calculated k_{eff}	0.1779
Standard Deviation	0.0004

† The maximum calculated k_{eff} of 0.1813 assumes an average ThO_2 content of 98.2 wt%, compared to the average ThO_2 content of 98.5 wt%. It is also based on a UO_2 content of 1.8 wt%. However, reducing the UO_2 content from 1.8 wt% to the average value of 1.5 wt% would result in a reduction of the already low reactivity, due to the reduction in the fissile material. Therefore the values listed in this table are more conservative.

6.3 MODEL SPECIFICATION

6.3.1 Description of Calculational Model

Figures 6.3.1 and 6.3.3 show representative horizontal cross sections of the three types of cells used in the calculations, and Figures 6.3.4 and 6.3.6 illustrate the basket configurations used. Three different MPC fuel basket designs were evaluated as follows:

- a 24 PWR assembly basket
- *a 32 PWR assembly basket*
- a 68 BWR assembly basket.

Full three-dimensional calculations were used, assuming the axial configuration shown in Figure 6.3.7, and conservatively neglecting the absorption in the overpack neutron shielding material (Holtite-A). Although the neutron absorber panels are 156 inches (396.24 cm) in length, which is much longer than the active fuel length with the maximum value of 150 inches (381 cm), they are assumed equal to the active fuel length in the calculations. As shown on the drawings in Section 1.5, 16 of the 24 periphery fixed neutron absorber panels on the MPC-24 have reduced width (i.e., 6.25 inches (15.875 cm) wide as opposed to 7.5 inches (19.05 cm)). However, the calculational models for the MPC-24 conservatively assume all of the periphery fixed neutron absorber panels are 6.25 inches (15.875 cm) in width.

The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 11 have no adverse effect on the design conditions important to criticality safety, and thus from a criticality standpoint, the normal, off-normal, and accident conditions are identical and do not require individual models.

The calculational model explicitly defines the fuel rods and cladding, the guide tubes (or water rods for BWR assemblies), the water-gaps and fixed neutron absorber panels on the stainless steel walls of the storage cells. Under the conditions of storage, when the MPC is dry, the resultant reactivity with the design basis fuel is very low ($k_{eff} < 0.4$). For the flooded condition (loading and unloading), water was assumed to be present in the fuel rod pellet-to-clad gaps. Appendix 6.D provides sample input files for two of the MPC basket designs *(MPC-68 and MPC-24)* in the HI-STAR 100 System.

The water thickness above and below the fuel is intentionally maintained less than or equal to the actual water thickness. This assures that any positive reactivity effect of the steel in the MPC is conservatively included. *Furthermore, the water above and below the fuel is modeled as unborated water, even when borated water is present in the fuel region.*

As indicated in Figures 6.3.1 and 6.3.3 and in Tables 6.3.1 and 6.3.2, calculations were made with dimensions assumed to be at their most conservative value with respect to criticality. CASMO-3 was used to determine the direction of the manufacturing tolerances which produced the most adverse effect on criticality. After the directional effect (positive effect with an increase in reactivity; or negative effect with a decrease in reactivity) of the manufacturing tolerances was determined, the criticality analyses were performed using the worst case tolerances in the direction which would increase reactivity. These effects are shown in Table 6.3.1 which also identifies the approximate magnitude of the tolerances on reactivity. *Generally, the conclusions in Table 6.3.1 are directly applicable to the MPC-32. Exceptions are the conclusions for the water temperature and void percentage, which are not directly applicable to the MPC-32 due to the presence of high soluble boron concentrations in this canister. This condition is addressed in Section 6.4.2.1 where the optimum moderation is determined for the MPC-32.*

The various basket dimensions are inter-dependent, and therefore cannot be individually varied (i.e., reduction in one parameter requires a corresponding reduction or increase in another parameter). Thus, it is not possible to determine the reactivity effect of each individual dimensional tolerance separately. However, it is possible to determine the reactivity effect of the dimensional tolerances by evaluating the various possible dimensional combinations. To this end, an evaluation of the various possible dimensional combinations was performed using MCNP4a. Calculated k_{eff} results (which do not include the bias, uncertainties, or calculational statistics), along with the actual dimensions, for a number of dimensional combinations are shown in Table 6.3.2 for the reference PWR and BWR assemblies. In Table 6.3.2, the box I.D. is the inner box dimension and the minimum, nominal, and maximum values correspond to those values permitted by the tolerances in the drawings in Section 1.5. *For PWR MPC designs, the reactivity effect of tolerances with soluble boron present in the water is additionally determined.*

Based on the MCNP4a and CASMO-3 calculations, the conservative dimensional assumptions listed in Table 6.3.3 were determined. Because the reactivity effect (positive or negative) of the manufacturing tolerances are not assembly dependent, these dimensional assumptions were employed for the criticality analyses.

As demonstrated in this section, design parameters important to criticality safety are: fuel enrichment, the inherent geometry of the fuel basket structure, and the fixed neutron absorbing panels. As shown in Chapter 11, none of these parameters are affected during any of the design basis off-normal or accident conditions involving handling, packaging, transfer or storage.

The MPC-32 criticality model uses a sheathing thickness of 0.075 inches (0.1905 cm), whereas the actual MPC-32 design uses a sheathing thickness of 0.035 inches (0.0889 cm). For the minimum cell pitch of 9.158 inches (23.26132 cm), the thicker sheathing results in a slightly smaller cell ID of 8.69 inches (22.0726 cm) (minimum), compared to 8.73 inches (22.1742 cm) (minimum) for the thinner sheathing. To demonstrate that the dimensions used in the criticality model are acceptable and conservative, calculations were performed for both sheathing *thicknesses and the results are compared in Table 6.3.5. To bound various soluble boron levels, two comparisons were performed. The first comparison uses the bounding case for the MPC-32 (see Table 6.1.6), which is for assembly class 15x15F at 5 wt% ^{235}U and a soluble boron level of 2600 ppm. To bound lower soluble boron levels, the second comparison uses the same assembly class (15x15F), 0 ppm soluble boron (i.e. pure water), and an arbitrary enrichment of 1.7 wt% ^{235}U. In both comparisons, the results of the 0.075 inch (0.1905 cm) sheathing are slightly higher, i.e. more conservative, than the results for 0.035 inch (0.0889 cm) sheathing, although the differences are within the statistical uncertainties. Using a sheathing thickness of 0.075 inches (0.1905 cm) in the criticality models of the MPC-32 is therefore acceptable, and potentially more conservative, than using the actual value of 0.035 inches (0.0889 cm). This validates the choice of the dimensional assumptions for the MPC-32 shown in Table 6.3.3, which are used for all further MPC-32 criticality calculations, unless otherwise noted.*

6.3.2 Cask Regional Densities

Composition of the various components of the principal designs of the HI-STAR 100 Systems are listed in Table 6.3.4.

The HI-STAR 100 System is designed such that the fixed neutron absorber will remain effective for a storage period greater than 20 years, and there are no credible means to lose it. A detailed physical description, historical applications, unique characteristics, service experience, and manufacturing quality assurance of fixed neutron absorber are provided in Section 1.2.1.3.1.

The continued efficacy of the fixed neutron absorber is assured by acceptance testing, documented in Section 9.1.5.3, to validate the ^{10}B (poison) concentration in the fixed neutron absorber. To demonstrate that the neutron flux from the irradiated fuel results in a negligible depletion of the poison material over the storage period, an MCNP4a calculation of the number of neutrons absorbed in the ^{10}B was performed. The calculation conservatively assumed a constant neutron source for 50 years equal to the initial source for the design basis fuel, as determined in Section 5.2, and shows that the fraction of ^{10}B atoms destroyed is only 2.6E-09 in 50 years. Thus, the reduction in ^{10}B concentration in the fixed neutron absorber by neutron absorption is negligible. In addition, analysis in Appendix 3.M.1 demonstrates that the sheathing, which affixes the fixed neutron absorber panel, remains in place during all credible accident conditions, and thus, the fixed neutron absorber panel remains permanently fixed. Therefore, in accordance with NUREG-1536, there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber, as required by 10CFR72.124(b).

6.3.3 Eccentric Positioning of Assemblies in Fuel Storage Cells

Up to and including Revision 3 of this FSAR, all criticality calculations were performed with fuel assemblies centered in the fuel storage locations since the effect of credible eccentric fuel *positioning was judged to be not significant. Starting in Revision 4 of this FSAR, the potential reactivity effect of eccentric positioning of assemblies in the fuel storage locations is accounted for in a conservatively bounding fashion, as described further in this subsection, for the new basket introduced, namely the MPC-32. The calculations in this subsection serve to determine for which of these conditions the eccentric positioning of assemblies in the fuel storage locations results in a higher maximum k_{eff} value than the centered positioning. For the cases where the eccentric positioning results in a higher maximum k_{eff} value, the eccentric positioning is used for all corresponding cases reported in the summary tables in Section 6.1 and the results tables in Section 6.4. All other calculations throughout this chapter, such as studies to determine bounding fuel dimensions, bounding basket dimensions, or bounding moderation conditions, are performed with assemblies centered in the fuel storage locations.*

To conservatively account for eccentric fuel positioning in the fuel storage cells, three different configurations are analyzed, and the results are compared to determine the bounding configuration:

- *Cell Center Configuration: All assemblies centered in their fuel storage cell; same configuration that is used in Section 6.2 and Section 6.3.1;*
- *Basket Center Configuration: All assemblies in the basket are moved as close to the center of the basket as permitted by the basket geometry; and*
- *Basket Periphery Configuration: All assemblies in the basket are moved furthest away from the basket center, and as close to the periphery of the basket as possible.*

It should be noted that the two eccentric configurations are hypothetical, since there is no known physical effect that could move all assemblies within a basket consistently to the center or periphery. Instead, the most likely configuration would be that all assemblies are moved in the same direction when the cask is in a horizontal position, and that assemblies are positioned randomly when the cask is in a vertical position. Further, it is not credible to assume that any such configuration could exist by chance. Even if the probability for a single assembly placed in the corner towards the basket center would be 1/5 (i.e. assuming only the center and four corner positions in each cell, all with equal probability), then the probability that all assemblies would be located towards the center would be $(1/5)^{24}$ or approximately 10^{-17} for the MPC-24, $(1/5)^{32}$ or approximately 10^{-23} for the MPC-32, and $(1/5)^{68}$ or approximately 10^{-48} for the MPC-68. However, since the configurations listed above bound all credible configurations, they are conservatively used in the analyses.

In Table 6.3.6, results are presented for the MPC-32 with soluble boron levels lower than 2600 ppm for 5 wt% ^{235}U and lower than 1900 ppm for 4.1 wt% ^{235}U. The table shows the maximum k_{eff} value for centered and the two eccentric configurations for each condition, and the difference in k_{eff} between the centered and eccentric positioning. The results and conclusions are summarized as follows:

- *In all cases, moving the assemblies to the periphery of the basket results in a reduction in reactivity, compared to the cell centered position.*
- *For the MPC-32 cases listed in Table 6.3.6, the maximum reactivity is shown for the basket center configuration. However, both the cell centered and basket centered configuration are analyzed for the MPC-32 design basis calculation, and the higher results are listed in the tables in Section 6.1 and 6.4. This applies to the cases with soluble boron levels lower than 2600 ppm for 5 wt% ^{235}U and lower than 1900 ppm for 4.1 wt% ^{235}U.*

Table 6.3.1

CASMO-3 CALCULATIONS FOR EFFECT OF TOLERANCES AND TEMPERATURE

Change in Nominal Parameter†	Δk for Maximum Tolerance		Action/Modeling Assumption
	MPC-24‡	MPC-68	
Reduce Fixed Neutron Absorber Width to Minimum	N/A††† min.= nom.= 7.5" and 6.25" (19.05 cm and 15.875 cm)	N/A††† min. = nom. = 4.75" (12.065 cm)	Assume minimum fixed neutron absorber width
Increase UO_2 Density to Maximum	+0.0017 max. = 10.522 g/cc nom. = 10.412 g/cc	+0.0014 max. = 10.522 g/cc nom. = 10.412 g/cc	Assume maximum UO_2 density
Reduce Box Inside Dimension (I.D.) to Minimum	-0.0005 min.= 8.86" (22.504 cm)nom. = 8.92" (22.657 cm)	See Table 6.3.2	Assume maximum box I.D. for the MPC-24
Increase Box Inside Dimension (I.D.) to Maximum	+0.0007 max. = 8.98" (22.809 cm) nom. = 8.92" (22.657 cm)	-0.0030 max. = 6.113" (15.527 cm) nom. = 6.053" (15.375 cm)	Assume minimum box I.D. for the MPC-68
Decrease Water Gap to Minimum	+0.0069 min. = 1.09" (2.769 cm) nom. = 1.15" (2.921 cm)	N/A	Assume minimum water gap in the MPC-24

† Reduction (or increase) in a parameter indicates that the parameter is changed to its minimum (or maximum) value.

‡ Calculations for the MPC-24 were performed with CASMO-4 [6.3.1–6.3.3]

††† The fixed neutron absorber width for the MPC-68 is 4.75" +0.125", -0" (12.065 cm + 0.3175 cm, - 0 cm), The fixed neutron absorber widths for the MPC-24 are 7.5" +0.125", -0" (19.05 cm + 0.3175 cm, - 0 cm) and 6.25" +0.125", -0" (15.875 cm + 0.3175 cm, - 0 cm) (i.e., the nominal and minimum values are the same).

Table 6.3.1 (continued)

CASMO-3 CALCULATIONS FOR EFFECT OF TOLERANCES AND TEMPERATURE

Change in Nominal Parameter	Δk Maximum Tolerance		Action/Modeling Assumption
	MPC-24	MPC-68	
Increase in Temperature 20°C 40°C 70°C 100°C	 Ref. -0.0030 -0.0089 -0.0162	 Ref. -0.0039 -0.0136 -0.0193	Assume 20°C
10% Void in Moderator 20°C with no void 20°C 100°C	 Ref. -0.0251 -0.0412	 Ref. -0.0241 -0.0432	Assume no void
Removal of Flow Channel (BWR)	N/A	-0.0073	Assume flow channel present for MPC-68

Table 6.3.2

MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES†

Pitch	Box I.D.	Box Wall Thickness	MCNP4a Calculated k_{eff}
MPC-24†† (17x17A01 @ 4.0% Enrichment)			
nominal (10.906") (27.701 cm)	maximum (8.98") (22.809 cm)	nominal (5/16") (0.794 cm)	0.9325±0.0008†††
minimum (10.846") (27.549 cm)	nominal (8.92") (22.657 cm)	nominal (5/16") (0.794 cm)	0.9300±0.0008
nominal (10.906") (27.701 cm)	nom. - 0.04" (8.88") (22.555 cm)	nom. + 0.05" (0.3625") (0.921 cm)	0.9305±0.0007
MPC-68 (8x8C04 @ 4.2% Enrichment)			
minimum (6.43") (16.332 cm)	minimum (5.993") (15.222 cm)	nominal (1/4") (0.635 cm)	0.9307±0.0007
nominal (6.49") (16.485 cm)	nominal (6.053") (15.375 cm)	nominal (1/4") (0.635 cm)	0.9274±0.0007
maximum (6.55") (16.637 cm)	maximum (6.113") (15.527 cm)	nominal (1/4") (0.635 cm)	0.9272±0.0008
nom. + 0.05" (6.54") (16.612 cm)	nominal (6.053") (15.375 cm)	nom. + 0.05" (0.30") (0.762 cm)	0.9267±0.0007

Note: Values in parentheses are the actual value used.

† Tolerance for pitch and box I.D. are ± 0.06"(0.1524 cm).
Tolerance for box wall thickness is +0.05"(0.127 cm), -0.00" (0 cm).

†† All calculations for the MPC-24 assume minimum water gap thickness (1.09") (2.769 cm).

††† Numbers are 1σ statistical uncertainties.

Table 6.3.2 (continued)
MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES†

Pitch	**Box I.D.**	**Box Wall Thickness**	**MCNP4a Calculated k_{eff}**
MPC-24 (17x17A @ 5.0% Enrichment) 400 ppm soluble boron			
nominal (10.906") (27.701 cm)	*maximum (8.98") (22.809 cm)*	*nominal (5/16") (0.794 cm)*	*0.9236±0.0007*††
maximum (10.966") (27.854 cm)	*maximum (8.98") (22.809 cm)*	*nominal (5/16") (0.794 cm)*	*0.9176±0.0008*
Minimum (10.846") (27.549 cm)	*nominal (8.92") (22.657 cm)*	*nominal (5/16") (0.794 cm)*	*0.9227±0.0010*
minimum (10.846") (27.549 cm	*minimum (8.86") (22.504 cm)*	*nominal (5/16") (0.794 cm)*	*0.9159±0.0008*
nominal (10.906") (27.701 cm)	*nominal-0.04" (8.88") (22.555 cm)*	*nom.+0.05" (0.3625") (0.921 cm)*	*0.9232±0.0009*
nominal (10.906") (27.701 cm)	*nominal (8.92") (22.657 cm)*	*nominal (5/16") (0.794 cm)*	*0.9158±0.0007*
MPC-32 (17x17A @ 5.0% Enrichment) 2600 ppm soluble boron†††			
Minimum (9.158") (23.261 cm)	*minimum (8.69") (22.073 cm)*	*nominal (9/32") (0.714 cm)*	*0.9085±0.0007*
Nominal (9.218") (23.414cm)	*nominal (8.75") (22.225 cm)*	*nominal (9/32") (0.714 cm)*	*0.9028±0.0007*
Maximum (9.278") (23.566 cm)	*maximum (8.81") (22.377 cm)*	*nominal (9/32") (0.714 cm)*	*0.8996±0.0008*
nominal+0.05" (9.268") (23.541 cm)	*nominal (8.75") (22.225 cm)*	*nominal+0.05" (0.331") (0.841 cm)*	*0.9023±0.0008*
minimum+0.05" (9.208") (23.388 cm)	*minimum (8.69") (22.073 cm)*	*nominal+0.05" (0.331") (0.841 cm)*	*0.9065±0.0007*
maximum (9.278") (23.566 cm)	*Maximum-0.05" (8.76") (22.250 cm)*	*nominal+0.05" (0.331") (0.841 cm)*	*0.9030±0.0008*

Notes:
1. *Values in parentheses are the actual value used.*

† *Tolerance for pitch and box I.D. are ± 0.06" (0.1524 cm).*
Tolerance for box wall thickness is +0.05" (0.127 cm), -0.00" (0 cm).

†† *Numbers are 1σ statistical uncertainties.*

††† *for 0.075" (0.1905 cm) sheathing thickness. See Section 6.3.1 and Table 6.3.5 for reactivity effect of sheathing thickness.*

Table 6.3.3

BASKET DIMENSIONAL ASSUMPTIONS

Basket Type	Pitch	Box I.D.	Box Wall Thickness	Water-Gap Flux Trap
MPC-24	nominal (10.906") (27.701 cm)	maximum (8.98") (22.809 cm)	nominal (5/16") (0.794 cm)	minimum (1.09") (2.769 cm)
MPC-32	*Minimum* *(9.158")* *(23.261 cm)*	*Minimum*† *(8.69")* *(22.073 cm)*	*Nominal* *(9/32")* *(0.714 cm)*	*N/A*
MPC-68	minimum (6.43") (16.332 cm)	minimum (5.993") (15.222 cm)	nominal (1/4") (0.635 cm)	N/A

† *for 0.075" (0.1905 cm) sheathing thickness. See Section 6.3.1 and Table 6.3.5 for reactivity effect of sheathing thickness.*

Table 6.3.4

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STAR 100 SYSTEM

MPC-24 and MPC-32		
UO_2 5.0% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	***Atom-Density***	***Wgt. Fraction***
8016	*4.696E-02*	*1.185E-01*
92235	*1.188E-03*	*4.408E-02*
92238	*2.229E-02*	*8.374E-01*
UO_2 4.0% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.693E-02	1.185E-01
92235	9.505E-04	3.526E-02
92238	2.252E-02	8.462E-01
BORAL (0.02 g ^{10}B/cm sq), DENSITY (g/cc) = 2.660 (MPC-24)		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	8.707E-03	5.443E-02
5011	3.512E-02	2.414E-01
6012	1.095E-02	8.210E-02
13027	3.694E-02	6.222E-01
BORAL (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.660 (MPC-32)		
Nuclide	***Atom-Density***	***Wgt. Fraction***
5010	*8.071E-03*	*5.089E-02*
5011	*3.255E-02*	*2.257E-01*
6012	*1.015E-02*	*7.675E-02*
13027	*3.805E-02*	*6.467E-01*

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STAR 100 SYSTEM

METAMIC (0.02 g ^{10}B/cm sq), DENSITY (g/cc) = 2.648 (MPC-24)		
Nuclide	***Atom-Density***	***Wgt. Fraction***
5010	*6.314E-03*	*3.965E-02*
5011	*2.542E-02*	*1.755E-01*
6012	*7.932E-02*	*5.975E-02*
13027	*4.286E-02*	*7.251E-01*
METAMIC (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.646 (MPC-32)		
Nuclide	***Atom-Density***	***Wgt. Fraction***
5010	*6.541E-03*	*4.110E-02*
5011	*2.633E-02*	*1.819E-01*
6012	*8.217E-03*	*6.193E-02*
13027	*4.223E-02*	*7.151E-01*

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STAR100 SYSTEM

BORATED WATER, 400 PPM, DENSITY (g/cc)=1.00		
Nuclide	***Atom-Density***	***Wgt. Fraction***
5010	4.330E-06	7.200E-05
5011	1.794E-05	3.280E-04
1001	6.683E-02	1.1185E-01
8016	3.341E-02	8.8775E-01
BORATED WATER, 1900 PPM, DENSITY (g/cc)=1.00		
Nuclide	***Atom-Density***	***Wgt. Fraction***
5010	*2.057E-05*	*3.420E-04*
5011	*8.522E-05*	*1.558E-03*
1001	*6.673E-02*	*1.1169E-01*
8016	*3.336E-02*	*8.8641E-01*
BORATED WATER, 2600 PPM, DENSITY (g/cc)=0.93		
Nuclide	***Atom-Density***	***Wgt. Fraction***
5010	*2.618e-05*	*4.680E-04*
5011	*1.085e-04*	*2.132E-03*
1001	*6.201e-02*	*1.1161E-01*
8016	*3.101e-02*	*8.8579E-01*

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STAR 100 SYSTEM

MPC-68		
UO_2 4.2% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.697E-02	1.185E-01
92235	9.983E-04	3.702E-02
92238	2.248E-02	8.445E-01
UO_2 3.0% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.695E-02	1.185E-01
92235	7.127E-04	2.644E-02
92238	2.276E-02	8.550E-01
MOX FUEL†, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.714E-02	1.190E-01
92235	1.719E-04	6.380E-03
92238	2.285E-02	8.584E-01
94239	3.876E-04	1.461E-02
94240	9.177E-06	3.400E-04
94241	3.247E-05	1.240E-03
94242	2.118E-06	7.000E-05

† The Pu-238, which is an absorber, was conservatively neglected in the MOX description for analysis purposes.

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STAR 100 SYSTEM

METAMIC (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.646		
Nuclide	***Atom-Density***	***Wgt. Fraction***
5010	*6.541E-03*	*4.110E-02*
5011	*2.633E-02*	*1.819E-01*
6012	*8.217E-03*	*6.193E-02*
13027	*4.223E-02*	*7.151E-01*
BORAL (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.660		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	8.071E-03	5.089E-02
5011	3.255E-02	2.257E-01
6012	1.015E-02	7.675E-02
13027	3.805E-02	6.467E-01
FUEL IN THORIA RODS, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.798E-02	1.212E-01
92235	4.001E-04	1.484E-02
92238	2.742E-05	1.030E-03
90232	2.357E-02	8.630E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STAR 100 SYSTEM

COMMON MATERIALS		
ZR CLAD, DENSITY (g/cc) = 6.550		
Nuclide	**Atom-Density**	**Wgt. Fraction**
40000	4.323E-02	1.000E+00
MODERATOR (H_2O), DENSITY (g/cc) = 1.000		
Nuclide	**Atom-Density**	**Wgt. Fraction**
1001	6.688E-02	1.119E-01
8016	3.344E-02	8.881E-01
STAINLESS STEEL, DENSITY (g/cc) = 7.840		
Nuclide	**Atom-Density**	**Wgt. Fraction**
24000	1.761E-02	1.894E-01
25055	1.761E-03	2.001E-02
26000	5.977E-02	6.905E-01
28000	8.239E-03	1.000E-01
ALUMINUM, DENSITY (g/cc) = 2.700		
Nuclide	**Atom-Density**	**Wgt. Fraction**
13027	6.026E-02	1.000E+00

Table 6.3.5

REACTIVITY EFFECT OF SHEATHING THICKNESS FOR THE MPC-32

Assembly Class	***Enrichment (wt% ^{235}U)***	***Soluble Boron (ppm)***	***Maximum k_{eff}***		***Difference in Maximum k_{eff}***
			Sheathing 0.075" (0.1095 cm) Min. Cell ID 8.69" (22.073 cm)	***Sheathing 0.035" (0.0889 cm) Min. Cell ID 8.73" (22.174 cm)***	
15x15F	*5.0*	*2600*	*0.9483*	*0.9476*	*-0.0008*
15x15F	*1.7*	*0*	*0.8914*	*0.8909*	*-0.0005*

Table 6.3.6

REACTIVITY EFFECTS OF ECCENTRIC POSITIONING OF CONTENT IN THE MPC-32

CASE	***Contents centered (Reference)***	***Content moved towards center of basket***		***Content moved towards basket periphery***	
	Maximum k_{eff}	***Maximum k_{eff}***	***k_{eff} Difference to Reference***	***Maximum k_{eff}***	***k_{eff} Difference to Reference***
MPC-32, Assembly Class 16x16A, 4.1% Enrichment, 1400 ppm Soluble Boron	*0.9332*	*0.9367*	*0.0035*	*0.8992*	*-0.0340*
MPC-32, Assembly Class 15x15B, 5.0% Enrichment, 2400 ppm Soluble Boron	*0.9473*	*0.9493*	*0.0020*	*0.9306*	*-0.0167*

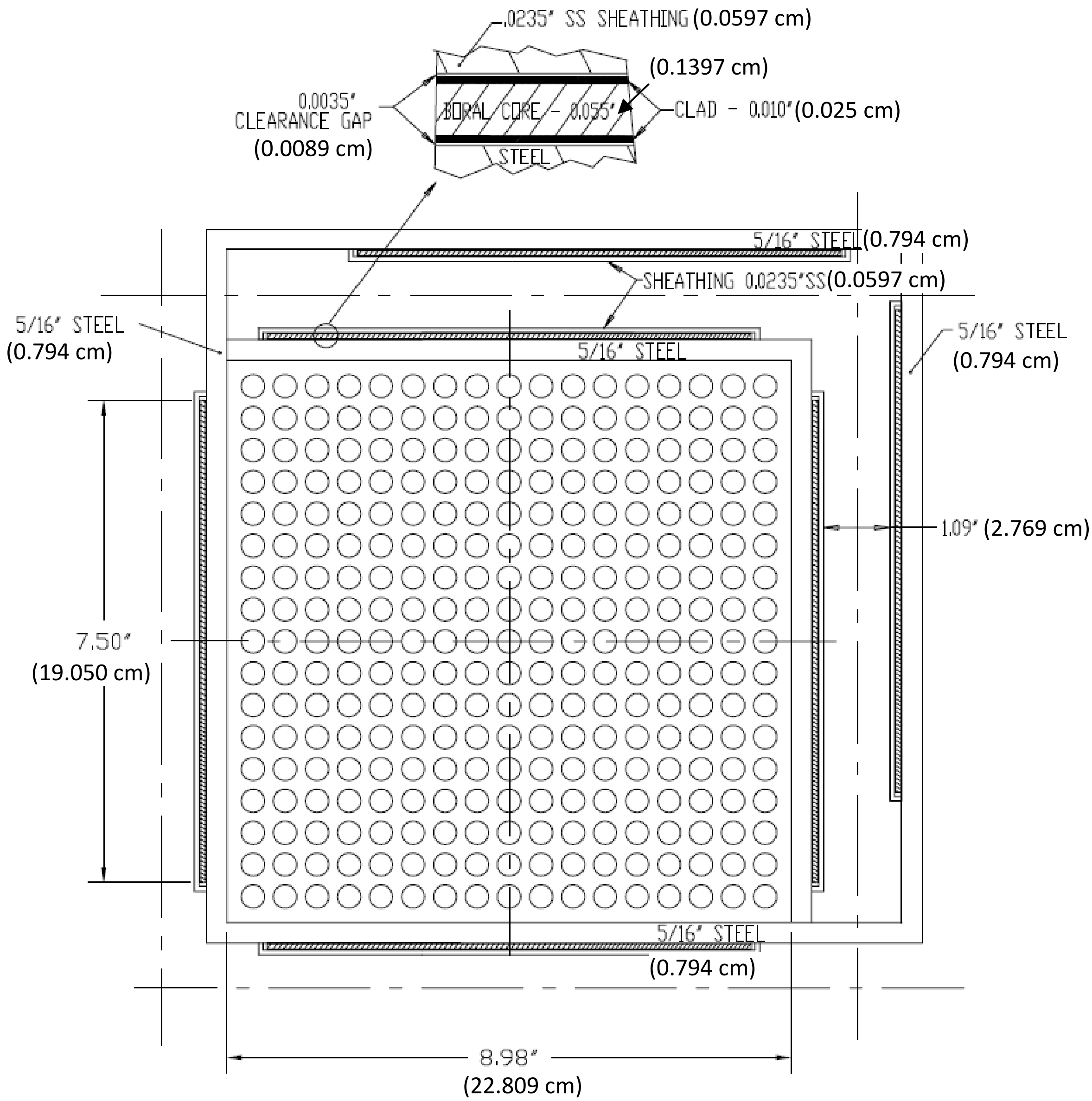

Figure 6.3.1 TYPICAL CELL IN THE CALCULATION MODEL (PLANAR CROSS-SECTION) WITH REPRESENTATIVE FUEL IN THE MPC-24 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

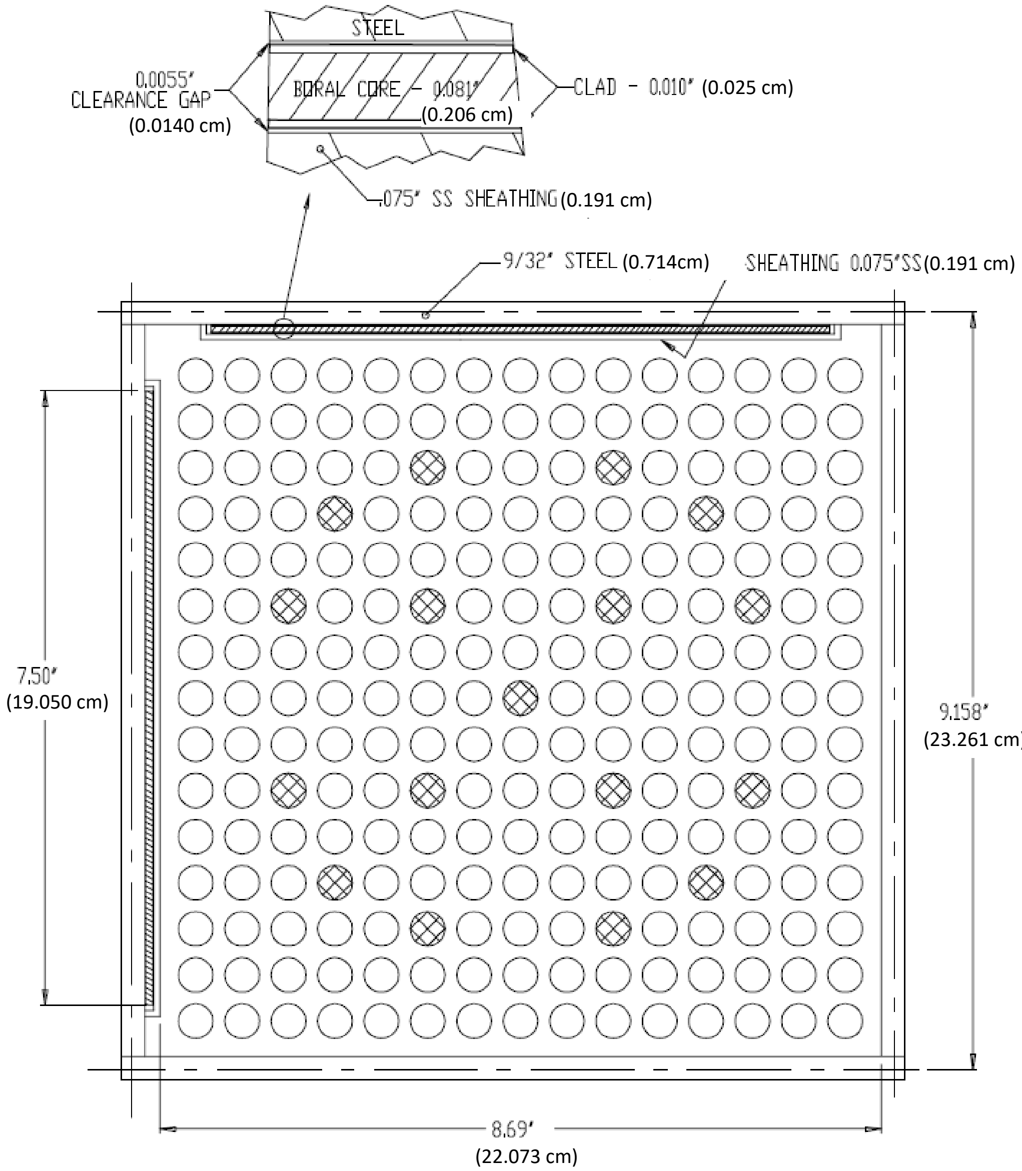

Figure 6.3.2 TYPICAL CELL IN THE CALCULATION MODEL (PLANAR CROSS-SECTION) WITH REPRESENTATIVE FUEL IN THE MPC-32 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

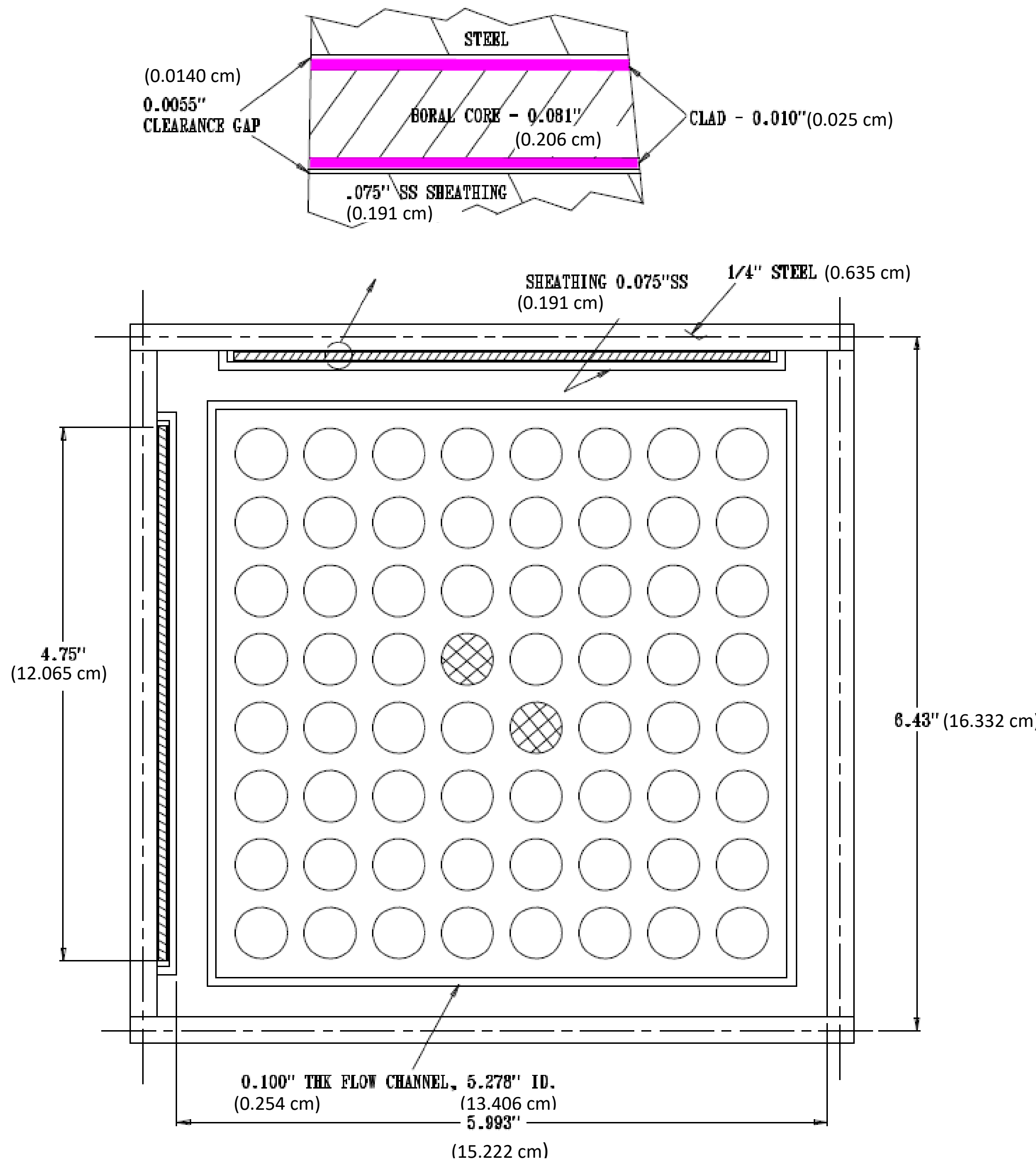

Figure 6.3.3 TYPICAL CELL IN THE CALCULATION MODEL (PLANAR CROSS-SECTION) WITH REPRESENTATIVE FUEL IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

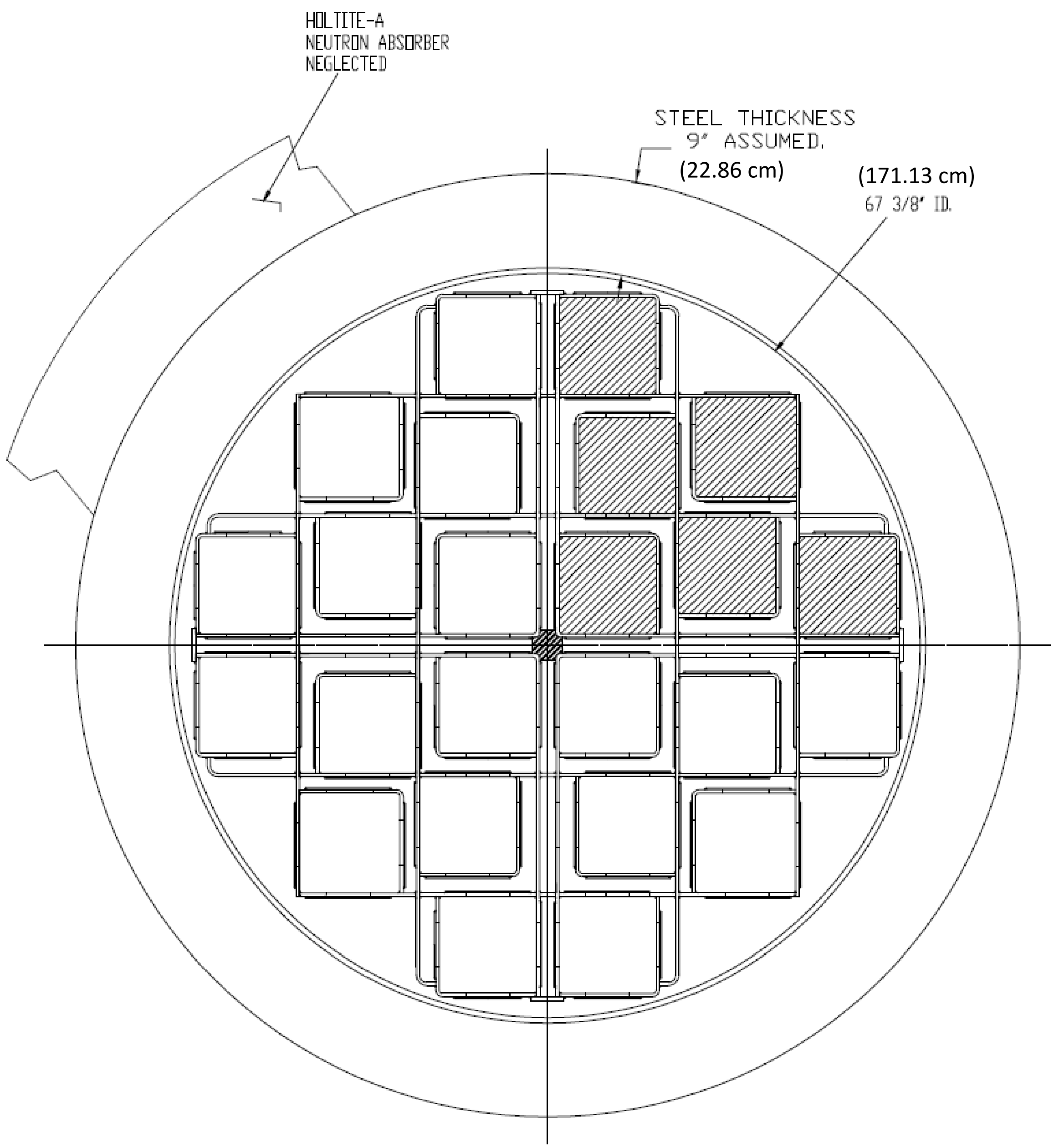

Figure 6.3.4 CALCULATION MODEL (PLANAR CROSS-SECTION) WITH FUEL ILLUSTRATED IN ONE QUADRANT OF THE MPC-24

(SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

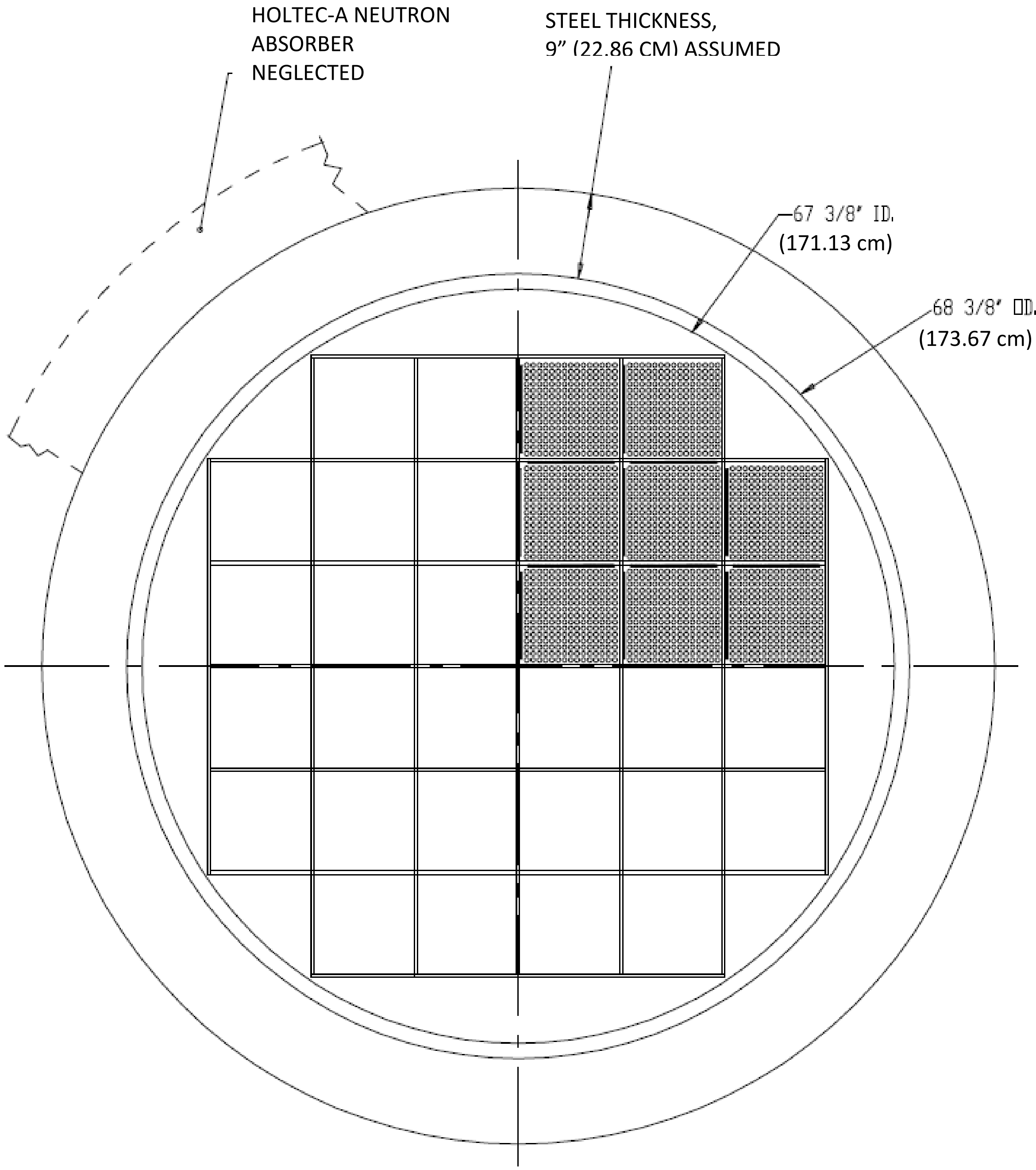

Figure 6.3.5 CALCULATION MODEL (PLANAR CROSS-SECTION) WITH FUEL ILLUSTRATED IN ONE QUADRANT OF THE MPC-32

(SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

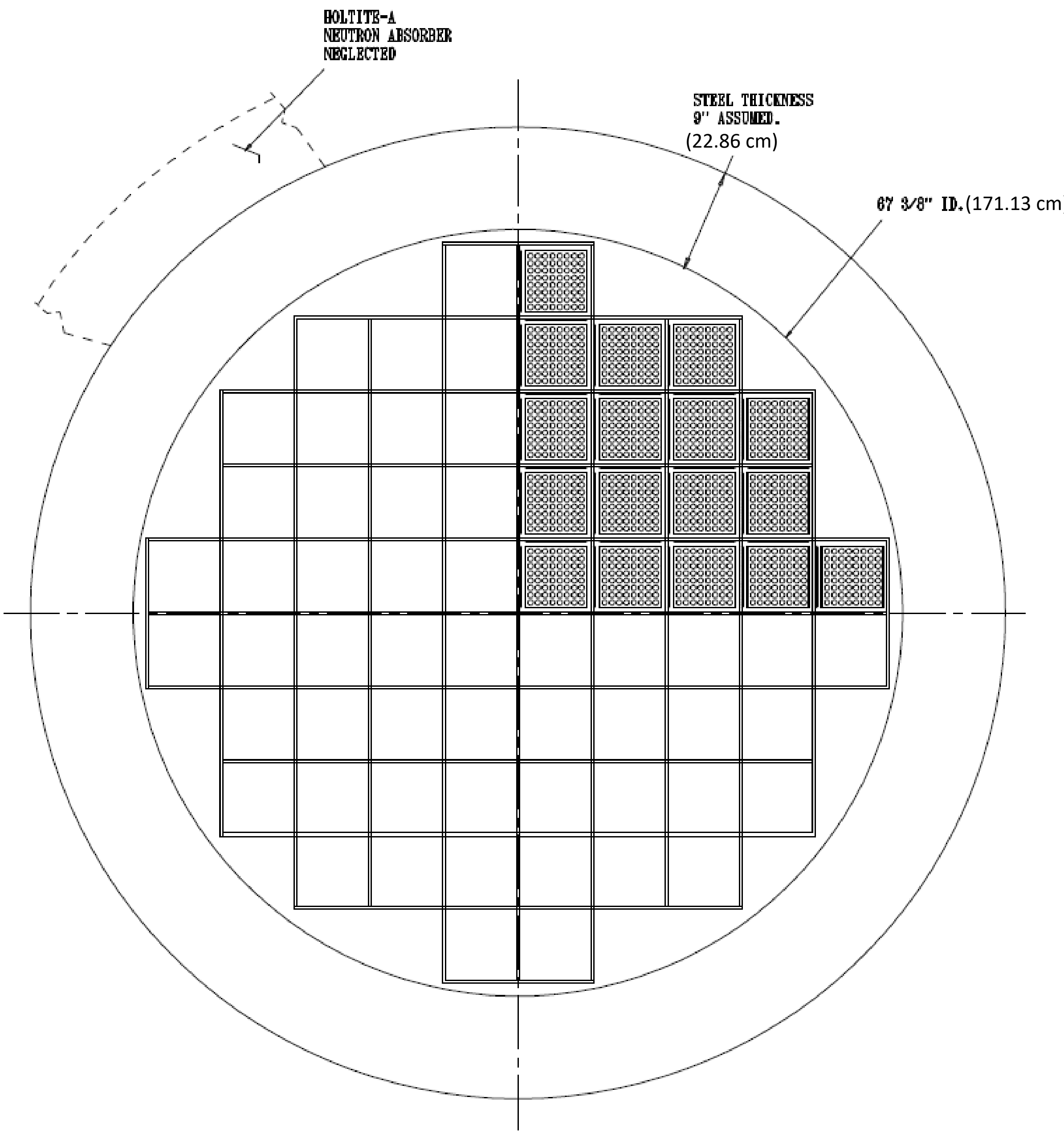

Figure 6.3.6 CALCULATION MODEL (PLANAR CROSS-SECTION) WITH FUEL ILLUSTRATED IN ONE QUADRANT OF THE MPC-68

(SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

	ACTIVE FUEL LENGTH	LOWER WATER THICKNESS	UPPER WATER THICKNESS
MPC-68	SEE TABLE 6.2.1	7.30 IN. (18.54 cm)	8.46 IN. (21.49 cm)
MPC-24 & MPC-32	SEE TABLE 6.2.2	4.0 IN. (10.16 cm)	6.0 IN. (15.24 cm)

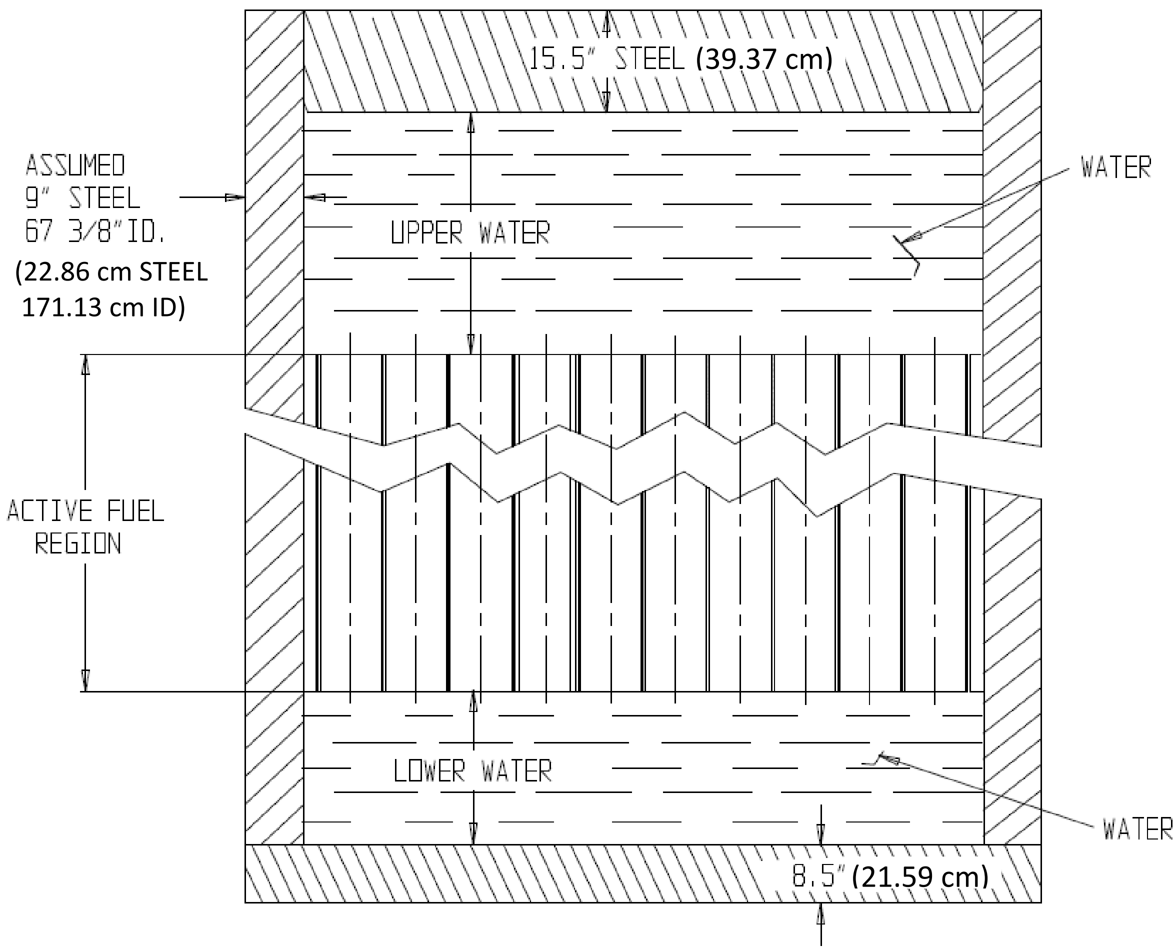

FIGURE 6.3.7 SKETCH OF THE CALCUALTIONAL MODEL IN THE AXIAL DIRECTION

6.4 CRITICALITY CALCULATIONS

6.4.1 Calculational or Experimental Method

6.4.1.1 Basic Criticality Safety Calculations

The principal method for the criticality analysis is the general three-dimensional continuous energy Monte Carlo N-Particle code MCNP4a [6.1.4] developed at the Los Alamos National Laboratory. MCNP4a was selected because it has been extensively used and verified and has all of the necessary features for this analysis. MCNP4a calculations used continuous energy cross-section data based on ENDF/B-V, as distributed with the code [6.1.4]. Independent verification calculations were performed with NITAWL-KENO5a [6.1.5], which is a three-dimensional multigroup Monte Carlo code developed at the Oak Ridge National Laboratory. The KENO5a calculations used the 238-group cross-section library, which is based on ENDF/B-V data and is distributed as part of the SCALE-4.3 package [6.4.1], compiled with the NITAWL-II program [6.1.6], which adjusts the uranium-238 cross sections to compensate for resonance self-shielding effects. The Dancoff factors required by NITAWL-II were calculated with the CELLDAN code [6.1.13], which includes the SUPERDAN code [6.1.7] as a subroutine.

The convergence of a Monte Carlo criticality problem is sensitive to the following parameters: (1) number of histories per cycle, (2) the number of cycles skipped before averaging, (3) the total number of cycles and (4) the initial source distribution. The MCNP4a criticality output contains a great deal of useful information that may be used to determine the acceptability of the problem convergence. This information was used in parametric studies to develop appropriate values for the aforementioned criticality parameters to be used in the criticality calculations for this submittal. Based on these studies, a minimum of 5,000 histories were simulated per cycle, a minimum of 20 cycles were skipped before averaging, a minimum of 100 cycles were accumulated, and the initial source was specified as uniform over the fueled regions (assemblies). Further, the output was examined to ensure that each calculation achieved acceptable convergence. These parameters represent an acceptable compromise between calculational precision and computational time. Appendix 6.D provides sample input files for *the MPC-24 and MPC-68 baskets in the HI-STAR 100 System.*

CASMO-3 [6.1.9] was used for determining the small incremental reactivity effects of manufacturing tolerances. Although CASMO-3 has been extensively benchmarked, these calculations are used only to establish direction of reactivity uncertainties due to manufacturing tolerances (and their magnitude). This allows the MCNP4a calculational model to use the worst combination of manufacturing tolerances. Table 6.3.1 shows results of the CASMO-3 calculations.

6.4.2 Fuel Loading or Other Contents Loading Optimization

The basket designs are intended to safely accommodate fuel with enrichments indicated in Tables 6.1.1 and 6.1.2. These calculations were based on the assumption that the HI-STAR 100 System was fully flooded with clean unborated water. In all cases, the calculations include bias and calculational uncertainties, as well as the reactivity effects of manufacturing tolerances, determined by assuming the worst case geometry.

6.4.2.1 Internal and External Moderation

As required by NUREG-1536, calculations in this section demonstrate that the HI-STAR 100 System remains subcritical for all credible conditions of moderation.

6.4.2.1.1 Unborated Water

With a neutron absorber present (i.e., the fixed neutron absorber sheets or the steel walls of the storage compartments), the phenomenon of a peak in reactivity at a hypothetical low moderator density (sometimes called "optimum" moderation) does not occur to any significant extent. In a definitive study, Cano, et al. [6.4.2] has demonstrated that the phenomenon of a peak in reactivity at low moderator densities does not occur when strong neutron absorbing material is present or in the absence of large water spaces between fuel assemblies in storage. Nevertheless, calculations for a single reflected cask were made to confirm that the phenomenon does not occur with low density water inside or outside the casks.

Calculations for the MPC designs with internal and external moderators of various densities are shown in Table 6.4.1. For comparison purposes, a calculation for a single unreflected cask (Case 1) is also included in Table 6.4.1. At 100% external moderator density, Case 2 corresponds to a single fully-flooded cask, fully reflected by water. Results listed in Table 6.4.1 support the following conclusions:

- For each type of MPC, the calculated k_{eff} for a fully-flooded cask is independent of the external moderator (the small variations in the listed values are due to statistical uncertainties which are inherent to the calculational method (Monte Carlo)), and

- For each type of MPC, reducing the internal moderation results in a monotonic reduction in reactivity, with no evidence of any optimum moderation. Thus, the fully flooded condition corresponds to the highest reactivity, and the phenomenon of optimum low-density moderation does not occur and is not applicable to the HI-STAR 100 System.

For each of the MPC designs, the maximum k_{eff} values are shown to be less than or statistically equal to that of a single internally flooded unreflected cask and are below the regulatory limit of

0.95.

6.4.2.1.2 Borated Water

With the presence of a soluble neutron absorber in the water, the discussion in the previous section is not always applicable. Calculations were made to determine the optimum moderator density for the MPC designs that require a minimum soluble boron concentration.

Calculations for the MPC designs with various internal moderator densities are shown in Table 6.4.6. As shown in the previous section, the external moderator density has a negligible effect on the reactivity, and is therefore not varied. Water containing soluble boron has a slightly higher density than pure water. Therefore, water densities up to 1.005 g/cm^3 were analyzed for the higher soluble boron concentrations. Additionally, for the higher soluble boron concentrations, analyses have been performed with empty (voided) guide tubes. This variation is discussed in detail in Section 6.4.8. Results listed in the Table 6.4.6 support the following conclusions:

- *For all cases with a soluble boron concentration of up to 1900 ppm, and for a soluble boron concentration of 2600 ppm assuming voided guide tubes, the conclusion of the Section 6.4.2.1.1 applies, i.e. the maximum reactivity is corresponds to 100% moderator density.*

- *For 2600 ppm soluble boron concentration with filled guide tubes, the results presented in Table 6.4.6 indicate that there is a maximum of the reactivity somewhere between 0.90 g/cm^3 and 1.00 g/cm^3 moderator density. However, a distinct maximum cannot be identified, as the reactivities in this range are very close. For the purpose of the calculations with 2600 ppm soluble boron concentration, a moderator density of 0.93 g/cm^3 was chosen, which corresponds to the highest calculated reactivity listed in Table 6.4.6.*

The calculations documented in this chapter also use soluble boron concentrations other than 1900 ppm and 2600 ppm in the MPC-32. For the MPC-32, soluble boron concentrations between 1300 ppm and 2600 ppm are used. In order to determine the optimum moderation condition for each assembly class at the corresponding soluble boron level, evaluations are performed with filled and voided guide tubes, and for water densities of 1.0 g/cm^3 and 0.93 g/cm^3 for each class and enrichment level. Results for the MPC-32 are listed in Table 6.4.7 for an initial enrichment of 5.0 wt% ^{235}U and in Table 6.4.8 for an initial enrichment of 4.1 wt% ^{235}U. The highest value listed in these tables for each assembly class is listed as the bounding value in Section 6.1.

6.4.2.2 Partial Flooding

As required by NUREG-1536, calculations in this section address partial flooding in the HI-STAR 100 System and demonstrate that the fully flooded condition is the most reactive.

The reactivity changes during the flooding process were evaluated in both the vertical and horizontal positions for all MPC designs. For these calculations, the cask is partially filled (at various levels) with full density (1.0 g/cc) water and the reminder of the cask is filled with steam consisting of ordinary water at partial density (0.002 g/cc), as suggested in NUREG-1536. Results of these calculations are shown in Table 6.4.2. In all cases, the reactivity increases monotonically as the water level rises, confirming that the most reactive condition is fully flooded.

6.4.2.3 Clad Gap Flooding

As required by NUREG-1536, the reactivity effect of flooding the fuel rod pellet-to-clad gap regions, in the fully flooded condition, has been investigated. Table 6.4.3 presents maximum k_{eff} values that demonstrate the positive reactivity effect associated with flooding the pellet-to-clad gap regions. These results confirm that it is conservative to assume that the pellet-to-clad gap regions are flooded. For all cases that involve flooding, the pellet-to-clad gap regions are assumed to be flooded.

6.4.2.4 Preferential Flooding

Preferential or uneven flooding within the HI-STAR 100 System was not evaluated because such a condition is not credible for any of the MPC basket designs loaded in the HI-STAR cask. Preferential flooding of any of the MPC fuel basket designs is not possible because flow holes are present on all four walls of each basket cell and on the two flux trap walls at both the top and bottom of the MPC basket. The flow holes are sized to ensure that they cannot be blocked by crud deposits (see Chapter 11). Because the fuel cladding temperatures remain below their design limits (as demonstrated in Chapter 4) and the inertial loading remains below 63g's (the inertial loadings associated with the design basis drop accidents discussed in Chapter 11 are limited to 60g's), the cladding remains intact (see Section 3.5). For damaged BWR fuel assemblies and BWR fuel debris, the assemblies or debris are pre-loaded into stainless steel Damaged Fuel Containers fitted with 250 micron fine mesh screens which prevent damaged fuel assemblies or fuel debris from blocking the basket flow holes. Therefore, the flow holes cannot be blocked.

Once established, the integrity of the MPC confinement boundary is maintained during all credible off-normal and accident conditions, and thus, the MPC cannot be flooded. Therefore, it is concluded that the MPC fuel baskets cannot be preferentially flooded.

6.4.2.5 Design Basis Accidents

The analyses presented in Chapters 3 and 11 demonstrate that the damage resulting from the design basis accidents is limited to a loss of the neutron shield material as a result of the fire accident. Because the criticality analyses do not take credit for the neutron shield material (Holtite-A), this condition has no effect on the criticality analyses.

As reported in Chapter 3, the minimum factor of safety for the MPC-24 as a result of the hypothetical cask drop or tip-over accident is 1.17 against the Level D allowables for Subsection NG, Section III of the ASME Code. Therefore, because the maximum box wall stresses are well within the ASME Level D allowables, the flux-trap gap change will be insignificant compared to the characteristic dimension of the flux trap.

In summary, the design basis accidents have no adverse effect on the design parameters important to criticality safety, and therefore, there is no increase in reactivity as a result of any of the credible off-normal or accident conditions involving handling, packaging, transfer or storage. Consequently, the HI-STAR 100 System is in full compliance with the requirement of 10CRF72.124, which states that “before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety.”

6.4.3 Criticality Results

Results of the criticality safety calculations for the condition of flooding with clean unborated water are presented in Section 6.2 and summarized in Section 6.1. These data confirm that for each of the candidate fuel types and basket configurations the effective multiplication factor (k_{eff}), including all biases and uncertainties at a 95-percent confidence level, do not exceed 0.95 under all credible normal, off-normal, and accident conditions.

Additional calculations (CASMO-3) at elevated temperatures confirm that the temperature coefficients of reactivity are negative as shown in Table 6.3.1. This confirms that the calculations for the storage baskets are conservative.

In calculating the maximum reactivity, the analysis used the following equation:

$$k_{eff}^{\max} = k_c + K_c \sigma_c + Bias + \sigma_B$$

where:

⇒ k_c is the calculated k_{eff} under the worst combination of tolerances;

⇒ K_c is the K multiplier for a one-sided statistical tolerance limit with 95% probability at the 95% confidence level [6.1.8]. Each final k_{eff} value calculated by MCNP4a (or KENO5a) is the result of averaging 100 (or more) cycle k_{eff} values, and thus, is based on a sample size of 100. The K multiplier corresponding to a sample size of 100 is 1.93. However, for this analysis a value of 2.00 was assumed for the K multiplier, which is larger (more conservative) than the value corresponding to a sample size of 100;

⇒ σ_c is the standard deviation of the calculated k_{eff}, as determined by the computer code (MCNP4a or KENO5a);

⇒ ***Bias*** is the systematic error in the calculations (code dependent) determined by comparison with critical experiments in Appendix 6.A; and

⇒ σ_B is the standard error of the bias (which includes the K multiplier for 95% probability at the 95% confidence level; see Appendix 6.A).

Appendix 6.A presents the critical experiment benchmarking and the derivation of the bias and standard error of the bias (95% probability at the 95% confidence level).

6.4.4 Damaged Fuel Container

Both damaged BWR fuel assemblies and BWR fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. Two different DFC types with slightly different cross sections are analyzed. DFCs containing fuel debris must be stored in the MPC-68F. DFCs containing damaged fuel assemblies may be stored in either the MPC-68 or MPC-68F. Evaluation of the capability of storing damaged fuel and fuel debris (loaded in DFCs) is limited to very low reactivity fuel in the MPC-68F. Because the MPC-68 has a higher specified ^{10}B loading, the evaluation of the MPC-68F conservatively bounds the storage of damaged BWR fuel assemblies in a standard MPC-68 Although the maximum planar-average enrichment of the damaged fuel is limited to 2.7% ^{235}U as specified in Appendix B to the Certificate of Compliance, analyses have been made for three possible scenarios, conservatively assuming fuel[††] of 3.0% enrichment. The scenarios considered included the following:

1. Lost or missing fuel rods, calculated for various numbers of missing rods in order to determine the maximum reactivity. The configurations assumed for analysis are illustrated in Figures 6.4.2 through 6.4.8.
2. Broken fuel assembly with the upper segments falling into the lower segment creating a close-packed array (described as a 8x8 array). For conservatism, the array analytically retained the same length as the original fuel assemblies in this analysis. This configuration is illustrated in Figure 6.4.9.
3. Fuel pellets lost from the assembly and forming powdered fuel dispersed through

[††] 6x6A01 and 7x7A01 fuel assemblies were used as representative assemblies.

a volume equivalent to the height of the original fuel. (Flow channel and clad material assumed to disappear).

Results of the analyses, shown in Table 6.4.5, confirm that, in all cases, the maximum reactivity is well below the regulatory limit. There is no significant difference in reactivity between the two DFC types. Collapsed fuel reactivity (simulating fuel debris) is low because of the reduced moderation. Dispersed powdered fuel results in low reactivity because of the increase in ^{238}U neutron capture (higher effective resonance integral for ^{238}U absorption).

The loss of fuel rods results in a small increase in reactivity (i.e., rods assumed to collapse, leaving a smaller number of rods still intact). The peak reactivity occurs for 8 missing rods, and a smaller (or larger) number of intact rods will have a lower reactivity, as indicated in Table 6.4.5.

The analyses performed and summarized in Table 6.4.5 provides the relative magnitude of the effects on the reactivity. This information coupled with the maximum k_{eff} values listed in Table 6.1.3 and the conservatism in the analyses, demonstrate that the maximum k_{eff} of the damaged fuel in the most adverse post-accident condition will remain well below the regulatory requirement of $k_{eff} < 0.95$.

Appendix 6.D provides sample input files for the damaged fuel analysis.

6.4.5 Fuel Assemblies with Missing Rods

For fuel assemblies that are qualified for damaged fuel storage, missing and/or damaged fuel rods are acceptable. However, for fuel assemblies to meet the limitations of intact fuel assembly storage, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods.

6.4.6 Thoria Rod Canister

The Thoria Rod Canister is similar to a DFC with an internal separator assembly containing 18 intact fuel rods. The configuration is illustrated in Figure 6.4.10. The k_{eff} value for an MPC-68F filled with Thoria Rod Canisters is calculated to be 0.1813. This low reactivity is attributed to the relatively low content in ^{235}U (equivalent to UO_2 fuel with an enrichment of approximately 1.7 wt% ^{235}U), the large spacing between the rods (the pitch is approximately 1" (2.54 cm), the cladding OD is 0.412" (1.046 cm)) and the absorption in the separator assembly. Together with the maximum k_{eff} values listed in Tables 6.1.2 and 6.1.3 this result demonstrates, that the k_{eff} for a Thoria Rod Canister loaded into the MPC-68 or the MPC-68F together with other approved fuel assemblies or DFCs will remain well below the regulatory requirement of $k_{eff} < 0.95$.

6.4.7 Sealed Rods replacing BWR Water Rods

Some BWR fuel assemblies contain sealed rods filled with a non-fissile instead of water rods. Compared to the configuration with water rods, the configuration with sealed rods has a reduced amount of moderator, while the amount of fissile material is maintained. Thus, the reactivity of the configuration with sealed rods will be lower compared to the configuration with water rods. Any configuration containing sealed rods instead of water rods is therefore bounded by the analysis for the configuration with water rods and no further analysis is required to demonstrate the acceptability. Therefore, for all BWR fuel assemblies analyzed, it is permissible that water rods are replaced by sealed rods filled with a non-fissile material.

6.4.8 Non-fuel Hardware in PWR Fuel Assemblies

Non-fuel Hardware in PWR fuel assemblies such as Thimble Plugs (TPs) and Burnable Poison Rod Assemblies (BPRAs) and similar devices are permitted for storage with all PWR fuel types. Non-fuel hardware is inserted in the guide tubes of the assemblies. For pure water, the reactivity of any PWR assembly with inserts is bounded by (i.e. lower than) the reactivity of the same assembly without the insert. This is due to the fact that the insert reduces the amount of moderator in the assembly, while the amount of fissile material remains unchanged.

With the presence of soluble boron in the water, non-fuel hardware not only displaces water, but also the neutron absorber in the water. It is therefore possible that the insertion results in an increase of reactivity, specifically for higher soluble boron concentrations. As a bounding approach for the presence of non-fuel hardware, analyses were performed with empty (voided) guide tubes, i.e. any absorption of the hardware is neglected. If assemblies contain an instrument tube, this tube remains filled with borated water. Table 6.4.6 shows results for the variation in water density for cases with filled and voided guide tubes. These results show that the optimum moderator density depends on the soluble boron concentration, and on whether the guide tubes are filled or assumed empty. For the MPC-24 with 400 ppm and the MPC-32 with 1900 ppm, voiding the guide tubes results in a reduction of reactivity. All calculations for the MPC-24 are therefore performed with water in the guide tubes. For the MPC-32 with 2600 ppm, the reactivity for voided guide tubes slightly exceeds the reactivity for filled guide tubes. However, this effect is not consistent across all assembly classes. Table 6.4.7 and Table 6.4.8 show results with filled and voided guide tubes for all assembly classes in the MPC-32 at 4.1 wt% ^{235}U and 5.0 wt% ^{235}U. Some classes show an increase, other classes show a decrease as a result of voiding the guide tubes. Therefore, for the results presented in the Section 6.1, Table 6.1.5 and Table 6.1.6, the maximum value for each class is chosen for each enrichment level.

An Instrument Tube Tie Rod (ITTR) is inserted into the instrument tube and is permitted for storage with all PWR assemblies. Studies for representative PWR assemblies, including the *assembly with the lowest margin (15x15F in the MPC-32), with voided instrument tubes confirm that this condition is equivalent to or bounded by the condition with flooded instrument tubes. An ITTR in the assembly instrument tube is therefore acceptable in all PWR assembly types, and all results and conclusions for PWR fuel assemblies without an ITTR are directly applicable to PWR assemblies with an ITTR.*

Therefore, from a criticality safety perspective, non-fuel hardware inserted into PWR assemblies are acceptable for all allowable PWR types, and, depending on the assembly class, can increase the safety margin.

6.4.9 Neutron Sources in Fuel Assemblies

Fuel assemblies containing start-up neutron sources are permitted for storage in the HI-STAR 100 System. The reactivity of a fuel assembly is not affected by the presence of a neutron source (other than by the presence of the material of the source, which is discussed later). This true because in a system with a k_{eff} less than 1.0, any given neutron population at any time, regardless of its origin or size, will decrease over time. Therefore, a neutron source of any strength will not increase reactivity, but only the neutron flux in a system, and no additional criticality analyses are required. Sources are inserted as rods into fuel assemblies, i.e. they replace either a fuel rod or water rod (moderator). Therefore, the insertion of the material of the source into a fuel assembly will not lead to an increase of reactivity either.

6.4.10 Fixed Neutron Absorber Material

The MPCs in the HI-STAR100 System can be manufactured with one of two possible neutron absorber materials: Boral or Metamic. Both materials are made of aluminum and B_4C powder. Boral has an inner core consisting of B_4C and aluminum between two outer layers consisting of aluminum only. This configuration is explicitly modeled in the criticality evaluation and shown in Figures 6.3.1 through 6.3.3 for each basket. Metamic is a single layer material with a slightly higher overall thickness and the same credited ^{10}B loading (in g/cm^2) for each basket. The majority of the criticality evaluations documented in this chapter are performed using Boral as the fixed neutron absorber. For a selected number of bounding cases, analyses are also performed using Metamic instead of Boral. (Note that the Metamic cases use the same absorber thickness as the corresponding Boral case, instead of the slightly increased thickness for Metamic. This is acceptable since analyses of slight thickness increases for a fixed ^{10}B loading (in g/cm^2) indicate that such increases have a negligible effect on reactivity.) The results for these cases are listed in Table 6.4.9, together with the corresponding result using Boral and the difference between the two materials for each case. Individual cases show small differences for the two materials. However, the differences are mostly below two times the standard deviation (the standard deviation is about 0.0008 for all cases in Table 6.4.9), indicating that the results *are statistically equivalent. Furthermore, all cases are below the regulatory limit of 0.95. Overall, the calculations demonstrate that the two fixed neutron absorber materials are identical from a criticality perspective. All results obtained for Boral are therefore directly applicable to Metamic and no further evaluations using Metamic are required.*

6.4.11 Annular Fuel Pellets

Typically, PWR fuel assemblies are designed with solid fuel pellets throughout the entire active fuel length. However, some PWR assemblies contain annular fuel pellets in the top and bottom 6 to 8 inches (15.24 to 20.32 cm) of the active fuel length. This changes the fuel to water ratio in these areas, which could have an effect on reactivity. However, the top and bottom of the active length are areas with high neutron leakage, and changes in these areas typically have no significant effect on reactivity. Studies with up to 12 inches (30.48 cm) of annular pellets at the top and bottom, with various pellet IDs confirm this, i.e., shown no significant reactivity effects, even if the annular region of the pellet is flooded with pure water. All calculations for PWR fuel assemblies are therefore performed with solid fuel pellets along the entire length of the active fuel region, and the results are directly applicable to those PWR assemblies with annular fuel pellets.

Table 6.4.1

MAXIMUM REACTIVITIES WITH REDUCED WATER DENSITIES FOR CASK ARRAYS†

	Water Density		**MCNP4a Maximum k_{eff}††**	
Case Number	**Internal**	**External**	**MPC-24 (17x17A01 @ 4.0%)**	**MPC-68 (8x8C04 @ 4.2%)**
1	100%	single cask	0.9368	0.9348
2	100%	100%	0.9354	0.9339
3	100%	70%	0.9362	0.9339
4	100%	50%	0.9352	0.9347
5	100%	20%	0.9372	0.9338
6	100%	10%	0.9380	0.9336
7	100%	5%	0.9351	0.9333
8	100%	0%	0.9342	0.9338
9	70%	0%	0.8337	0.8488
10	50%	0%	0.7426	0.7631
11	20%	0%	0.5606	0.5797
12	10%	0%	0.4834	0.5139
13	5%	0%	0.4432	0.4763
14	10%	100%	0.4793	0.4946

† For an infinite square array of casks with 60cm spacing between cask surfaces

†† Maximum k_{eff} includes the bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.2

REACTIVITY EFFECTS OF PARTIAL CASK FLOODING

MPC-24 (17x17A01 @ 4.0% ENRICHMENT) ***(no soluble boron)***			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.9157	25	0.8766
50	0.9305	50	0.9240
75	0.9330	75	0.9329
100	0.9368	100	0.9368
MPC-68 (8x8C04 @ 4.2% ENRICHMENT)			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.9132	23.5	0.8586
50	0.9307	50	0.9088
75	0.9312	76.5	0.9275
100	0.9348	100	0.9348
MPC-32 (15x15F @ 5.0 % ENRICHMENT) 2600ppm Soluble Boron			
Flooded Condition (% Full)	*Vertical Orientation*	*Flooded Condition (% Full)*	*Horizontal Orientation*
25	*0.8927*	*31.25*	*0.9213*
50	*0.9215*	*50*	*0.9388*
75	*0.9350*	*68.75*	*0.9401*
100	*0.9445*	*100*	*0.9445*

Notes:

1. All values are maximum k_{eff} which include bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.3

REACTIVITY EFFECT OF FLOODING THE PELLET-TO-CLAD GAP

Pellet-to-Clad Condition	MPC-24 17x17A01 4.0% Enrichment	MPC-68 8x8C04 4.2% Enrichment
dry	0.9295	0.9279
flooded	0.9368	0.9348

Notes:

1. All values are maximum k_{eff} which includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.4

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Table 6.4.5

MAXIMUM k_{eff} VALUES[†] IN THE DAMAGED FUEL CONTAINER

Condition	**MCNP4a Maximum[††] k_{eff}**	
	DFC Dimensions: ID 4.93" (12.522 cm) THK. 0.12" (0.305 cm)	**DFC Dimensions: ID 4.81" (12.217 cm) THK. 0.11" (0.279 cm)**
<u>6x6 Fuel Assembly</u>		
6x6 Intact Fuel	0.7086	0.7016
w/32 Rods Standing	0.7183	0.7117
w/28 Rods Standing	0.7315	0.7241
w/24 Rods Standing	0.7086	0.7010
w/18 Rods Standing	0.6524	0.6453
Collapsed to 8x8 array	0.7845	0.7857
Dispersed Powder	0.7628	0.7440
<u>7x7 Fuel Assembly</u>		
7x7 Intact Fuel	0.7463	0.7393
w/41 Rods Standing	0.7529	0.7481
w/36 Rods Standing	0.7487	0.7444
w/25 Rods Standing	0.6718	0.6644

† These calculations were performed with a planar-average enrichment of 3.0% and a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68F is 0.010 g/cm^2. Therefore, the listed maximum k_{eff} values are conservative.

†† Maximum k_{eff} includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.6

MAXIMUM k_{eff} VALUES WITH REDUCED BORATED WATER DENSITIES

Internal Water Density† in g/cm³	***Maximum k_{eff}***				
	MPC-24 (400ppm) @ 5.0 %	***MPC-32 (1900ppm) @ 4.1 %***		***MPC-32 (2600ppm) @ 5.0 %***	
Guide Tubes	*filled*	*filled*	*void*	*filled*	*void*
1.005	*NC††*	*0.9403*	*0.9395*	*NC*	*0.9481*
1.00	*0.9314*	*0.9411*	*0.9400*	*0.9445*	*0.9483*
0.99	*NC*	*0.9393*	*0.9396*	*0.9438*	*0.9462*
0.98	*0.9245*	*0.9403*	*0.9376*	*0.9447*	*0.9465*
0.97	*NC*	*0.9397*	*0.9391*	*0.9453*	*0.9476*
0.96	*NC*	*NC*	*NC*	*0.9446*	*0.9466*
0.95	*0.9186*	*0.9380*	*0.9384*	*0.9451*	*0.9468*
0.94	*NC*	*NC*	*NC*	*0.9445*	*0.9467*
0.93	*0.9130*	*0.9392*	*0.9352*	*0.9465*	*0.9460*
0.92	*NC*	*NC*	*NC*	*0.9458*	*0.9450*
0.91	*NC*	*NC*	*NC*	*0.9447*	*0.9452*
0.90	*0.9061*	*0.9384*	*NC*	*0.9449*	*0.9454*
0.80	*0.8774*	*0.9322*	*NC*	*0.9431*	*0.9390*
0.70	*0.8457*	*0.9190*	*NC*	*0.9339*	*0.9259*
0.60	*0.8095*	*0.8990*	*NC*	*0.9194*	*0.9058*
0.40	*0.7225*	*0.8280*	*NC*	*0.8575*	*0.8410*
0.20	*0.6131*	*0.7002*	*NC*	*0.7421*	*0.7271*
0.10	*0.5486*	*0.6178*	*NC*	*0.6662*	*0.6584*

† External moderator is modeled at 0%. This is consistent with the results demonstrated in Table 6.4.1.
†† NC: Not Calculated

Table 6.4.7

MAXIMUM k_{eff} VALUES WITH FILLED AND VOIDED GUIDE TUBES FOR THE MPC-32 AT 5.0 wt% ENRICHMENT

Fuel Class	***Minimum Soluble Boron Content (ppm)***	***MPC-32 @ 5.0 %***			
		Guide Tubes Filled,		***Guide Tubes Voided,***	
		1.0 g/cm^3	***0.93 g/cm^3***	***1.0 g/cm^3***	***0.93 g/cm^3***
14x14A	*1900*	*0.8984*	*0.9000*	*0.8953*	*0.8943*
14x14B	*1900*	*0.9210*	*0.9214*	*0.9164*	*0.9118*
14x14C	*1900*	*0.9371*	*0.9376*	*0.9480*	*0.9421*
14x14D	*1900*	*0.9050*	*0.9027*	*0.8947*	*0.8904*
15x15A	*2500*	*0.9210*	*0.9223*	*0.9230*	*0.9210*
15x15B	*2500*	*0.9402*	*0.9420*	*0.9429*	*0.9421*
15x15C	*2500*	*0.9258*	*0.9292*	*0.9307*	*0.9293*
15x15D	*2600*	*0.9426*	*0.9419*	*0.9466*	*0.9440*
15x15E	*2600*	*0.9394*	*0.9415*	*0.9434*	*0.9442*
15x15F	*2600*	*0.9445*	*0.9465*	*0.9483*	*0.9460*
15x15G	*2500*	*0.9228*	*0.9244*	*0.9251*	*0.9243*
15X15H	*2600*	*0.9271*	*0.9301*	*0.9317*	*0.9333*
16X16A	*2000*	*0.9377*	*0.9375*	*0.9429*	*0.9389*
17x17A	*2600*	*0.9105*	*0.9145*	*0.9160*	*0.9161*
17x17B	*2600*	*0.9345*	*0.9358*	*0.9371*	*0.9356*
17X17C	*2600*	*0.9417*	*0.9431*	*0.9437*	*0.9430*

Table 6.4.8

MAXIMUM k_{eff} VALUES WITH FILLED AND VOIDED GUIDE TUBES FOR THE MPC-32 AT 4.1 wt% ENRICHMENT

Fuel Class	Minimum Soluble Boron Content (ppm)	MPC-32 @ 4.1 %			
		Guide Tubes Filled		Guide Tubes Voided	
		1.0 g/cm^3	0.93 g/cm^3	1.0 g/cm^3	0.93 g/cm^3
14x14A	1300	0.9041	0.9029	0.8954	0.8939
14x14B	1300	0.9257	0.9205	0.9128	0.9074
14x14C	1300	0.9402	0.9384	0.9423	0.9365
14x14D	1300	0.8970	0.8943	0.8836	0.8788
15x15A	1800	0.9199	0.9206	0.9193	0.9134
15x15B	1800	0.9397	0.9387	0.9385	0.9347
15x15C	1800	0.9266	0.9250	0.9264	0.9236
15x15D	1900	0.9375	0.9384	0.9380	0.9329
15x15E	1900	0.9348	0.9340	0.9365	0.9336
15x15F	1900	0.9411	0.9392	0.9400	0.9352
15x15G	1800	0.9147	0.9128	0.9125	0.9062
15X15H	1900	0.9267	0.9274	0.9276	0.9268
16X16A	1400	0.9367	0.9347	0.9375	0.9308
17x17A	1900	0.9105	0.9111	0.9106	0.9091
17x17B	1900	0.9309	0.9307	0.9297	0.9243
17X17C	1900	0.9355	0.9347	0.9350	0.9308

Table 6.4.9

COMPARISON OF MAXIMUM k_{eff} VALUES FOR DIFFERENT FIXED NEUTRON ABSORBER MATERIALS

Case	***Maximum k_{eff}***		***Reactivity Difference***
	BORAL	***METAMIC***	
MPC-68, Intact Assemblies	*0.9457*	*0.9452*	*-0.0005*
MPC-68, with 16 DFCs	*0.9328*	*0.9315*	*-0.0013*
MPC-68F with 68 DFCs	*0.8021*	*0.8019*	*-0.0002*
MPC-24, 0ppm	*0.9478*	*0.9491*	*+0.0013*
MPC-24, 400ppm	*0.9447*	*0.9457*	*+0.0010*
MPC-32, 1900ppm	*0.9411*	*0.9397*	*-0.0014*
MPC-32, 2600ppm	*0.9483*	*0.9471*	*-0.0012*

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FIGURE 6.4.1

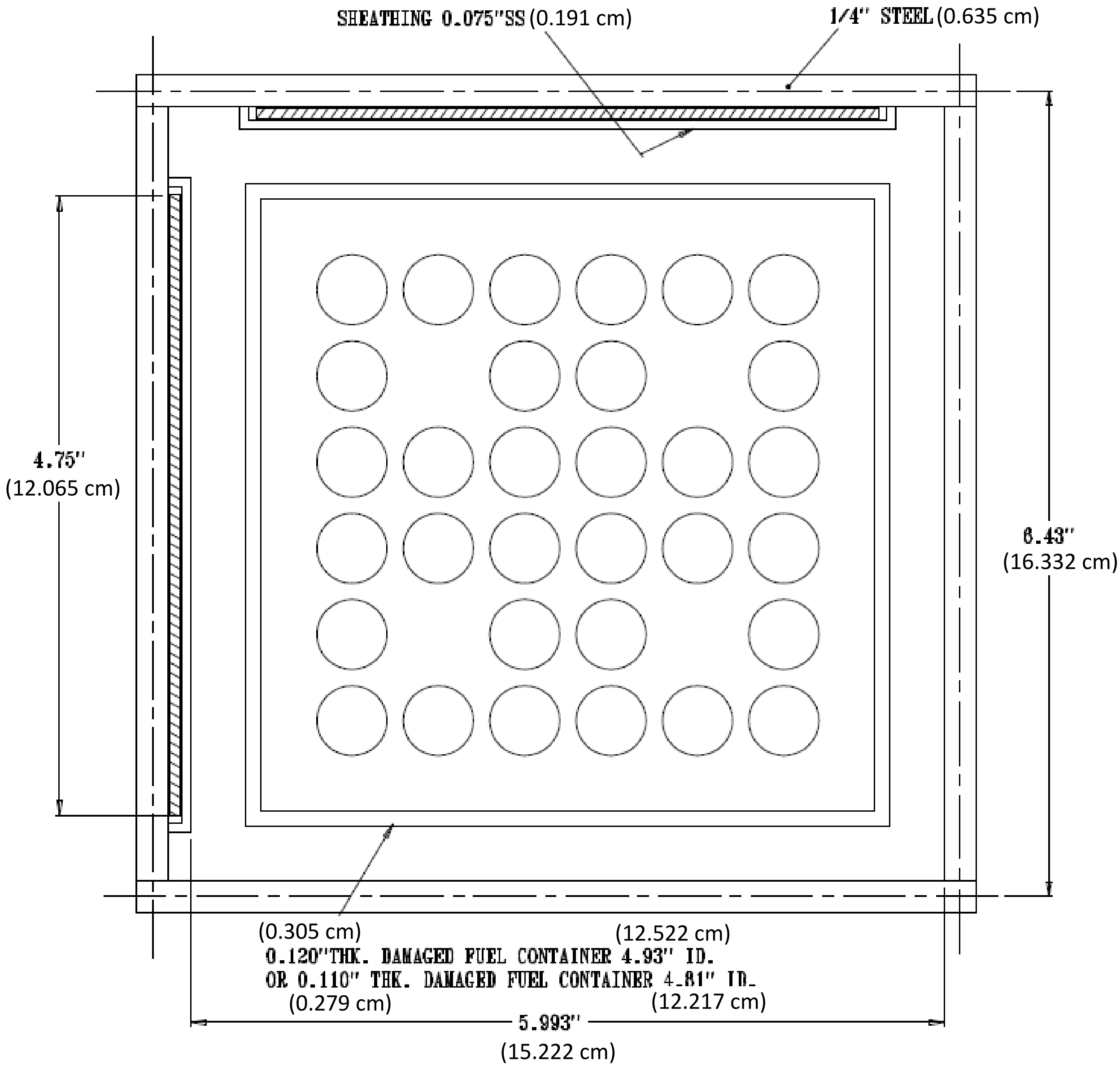

FIGURE 6.4.2 FAILED FUEL CALCUALTION MODEL (PLANAR CROSS-SECTION) WITH 6X6 ARRAY WITH 4 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

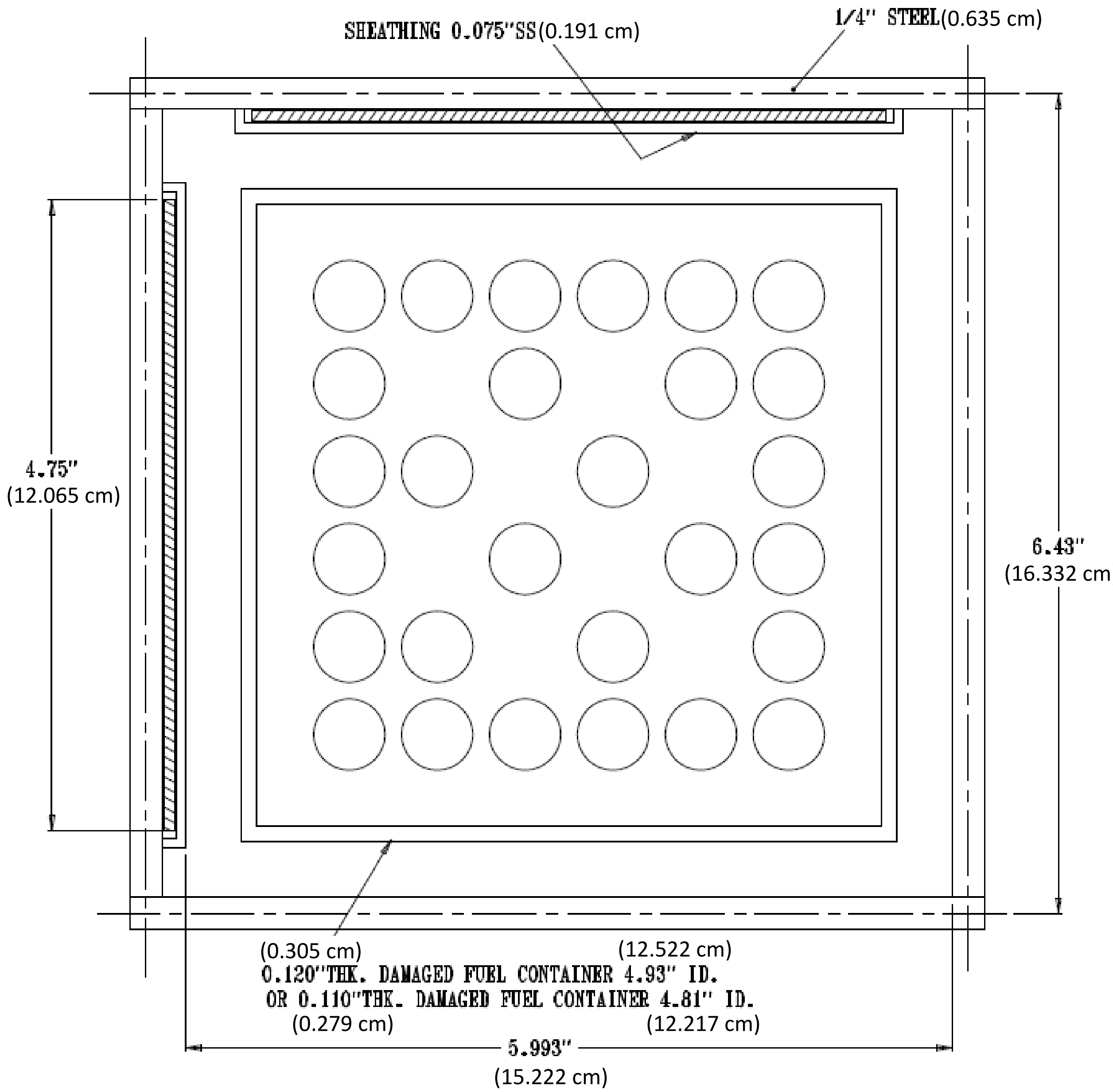

FIGURE 6.4.3 FAILED FUEL CALCUALTION MODEL (PLANAR CROSS-SECTION) WITH 6X6 ARRAY WITH 8 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

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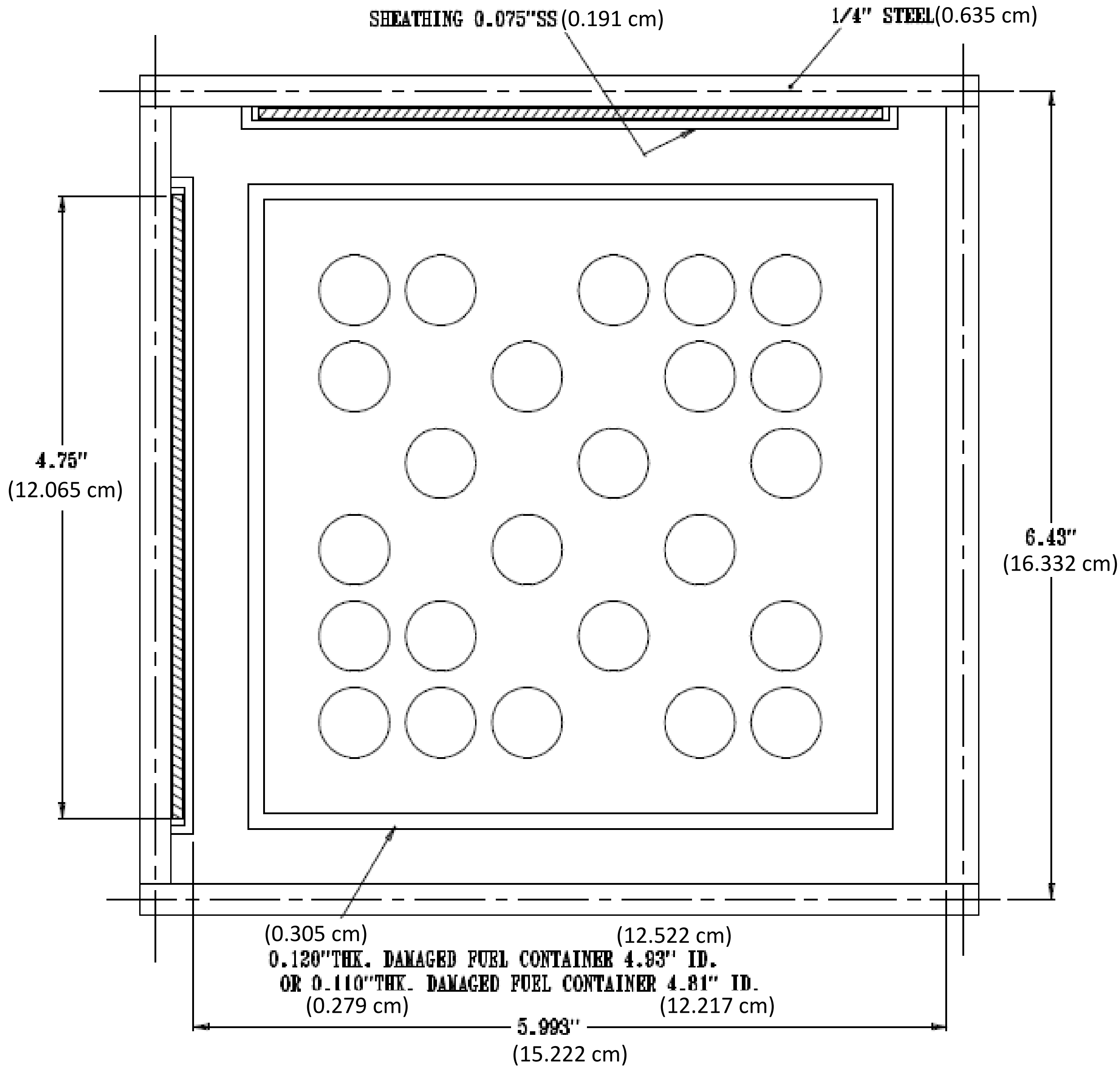

FIGURE 6.4.4 FAILED FUEL CALCUALTION MODEL (PLANAR CROSS-SECTION) WITH 6X6 ARRAY WITH 12 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

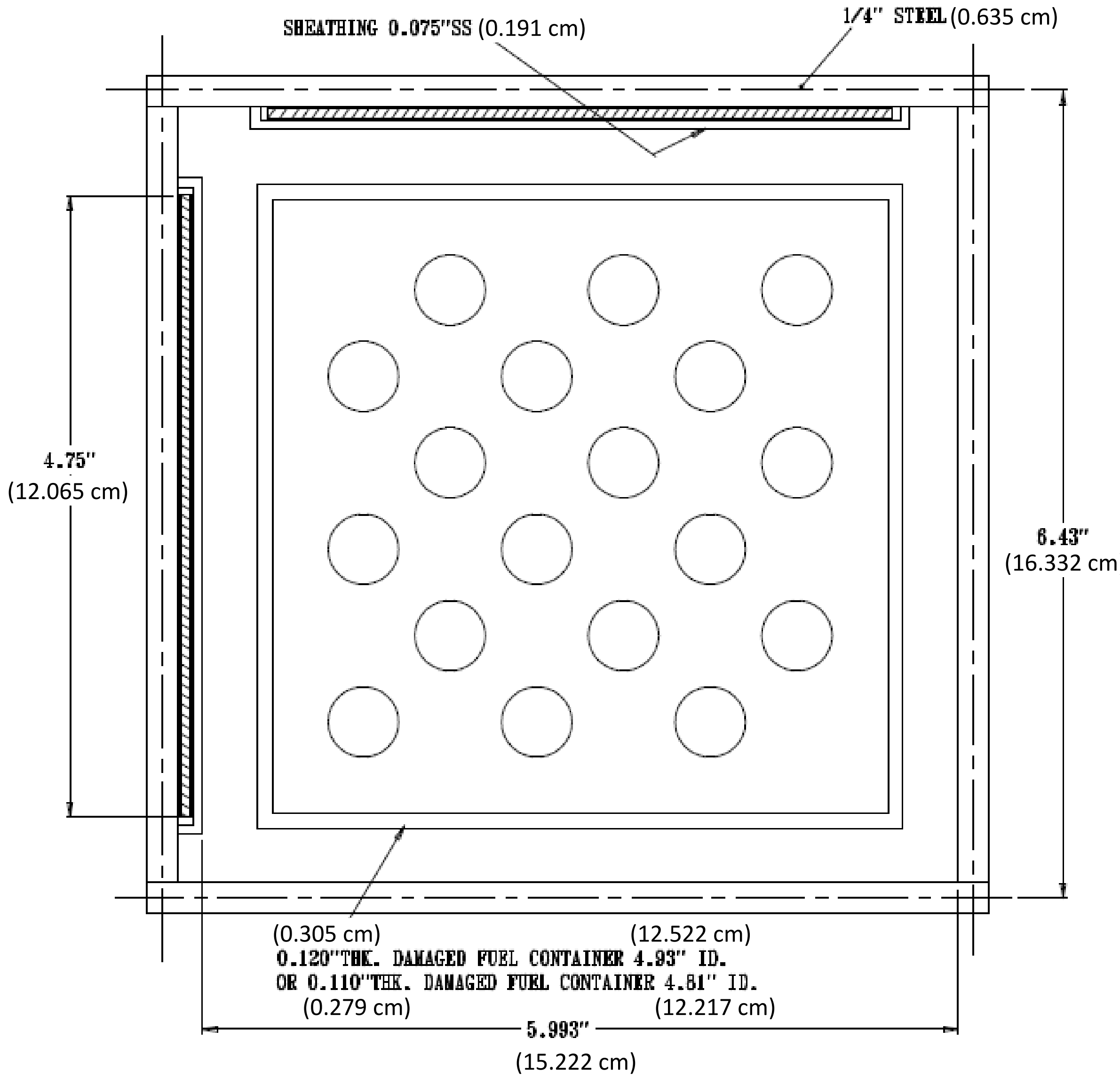

FIGURE 6.4.5 FAILED FUEL CALCUALTION MODEL (PLANAR CROSS-SECTION) WITH 6X6 ARRAY WITH 18 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

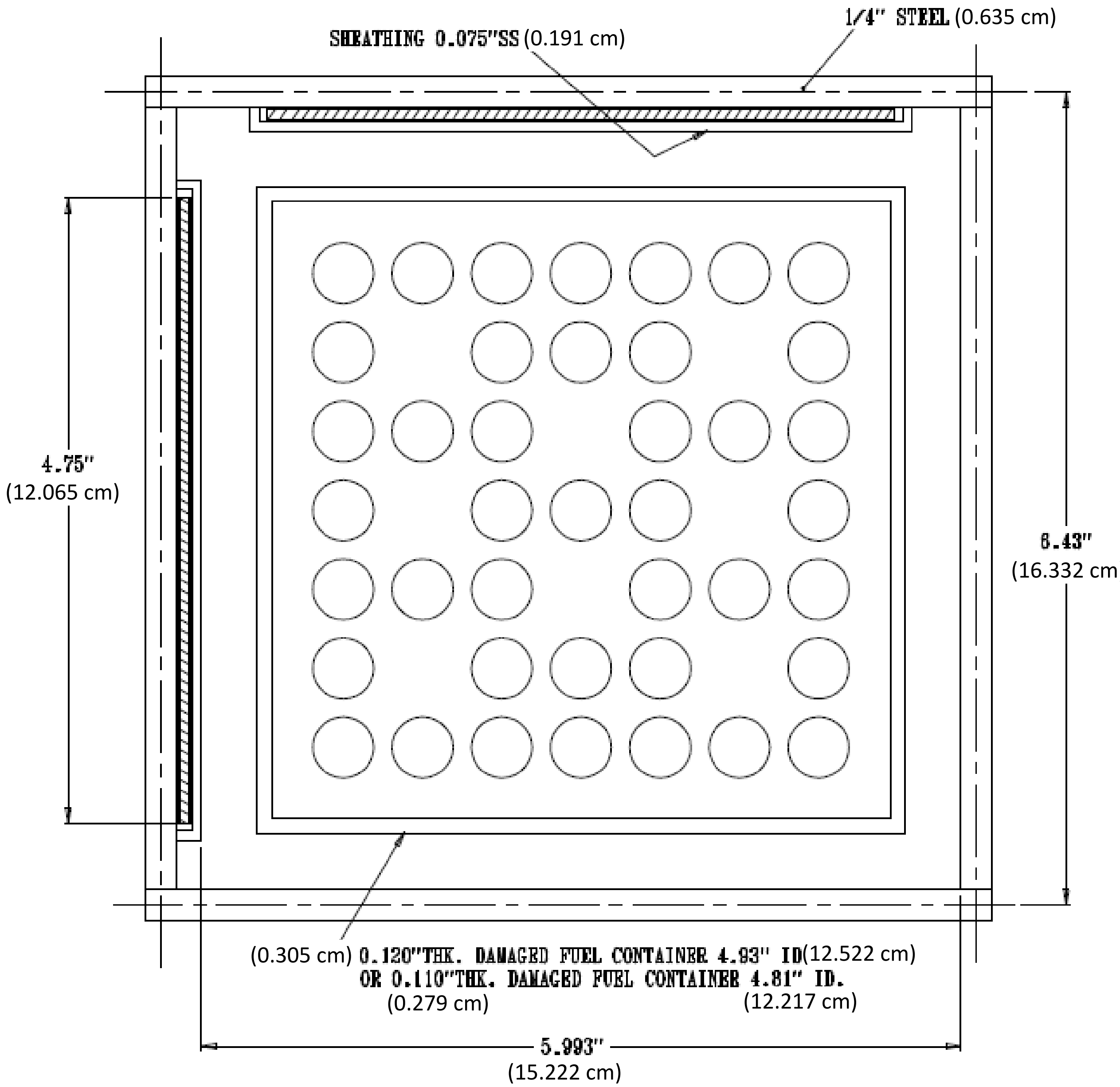

FIGURE 6.4.6 FAILED FUEL CALCUALTION MODEL (PLANAR CROSS-SECTION) WITH 7X7 ARRAY WITH 8 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

NOTE: THESE DIMENSIONS WERE CONSERVATIVELY USED FOR CRITICALITY ANALYSES.

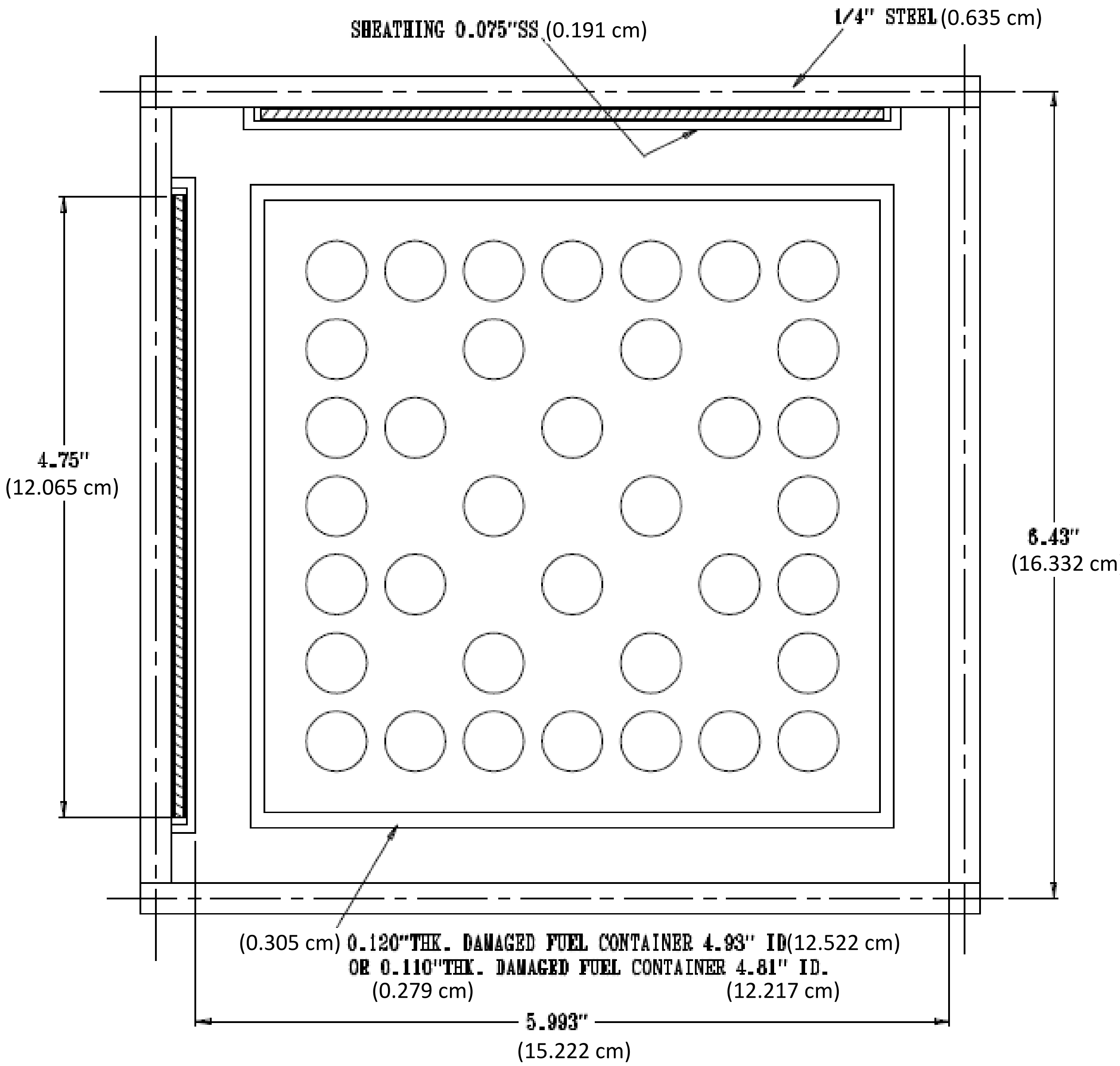

FIGURE 6.4.7 FAILED FUEL CALCUALTION MODEL (PLANAR CROSS-SECTION) WITH 7X7 ARRAY WITH 13 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

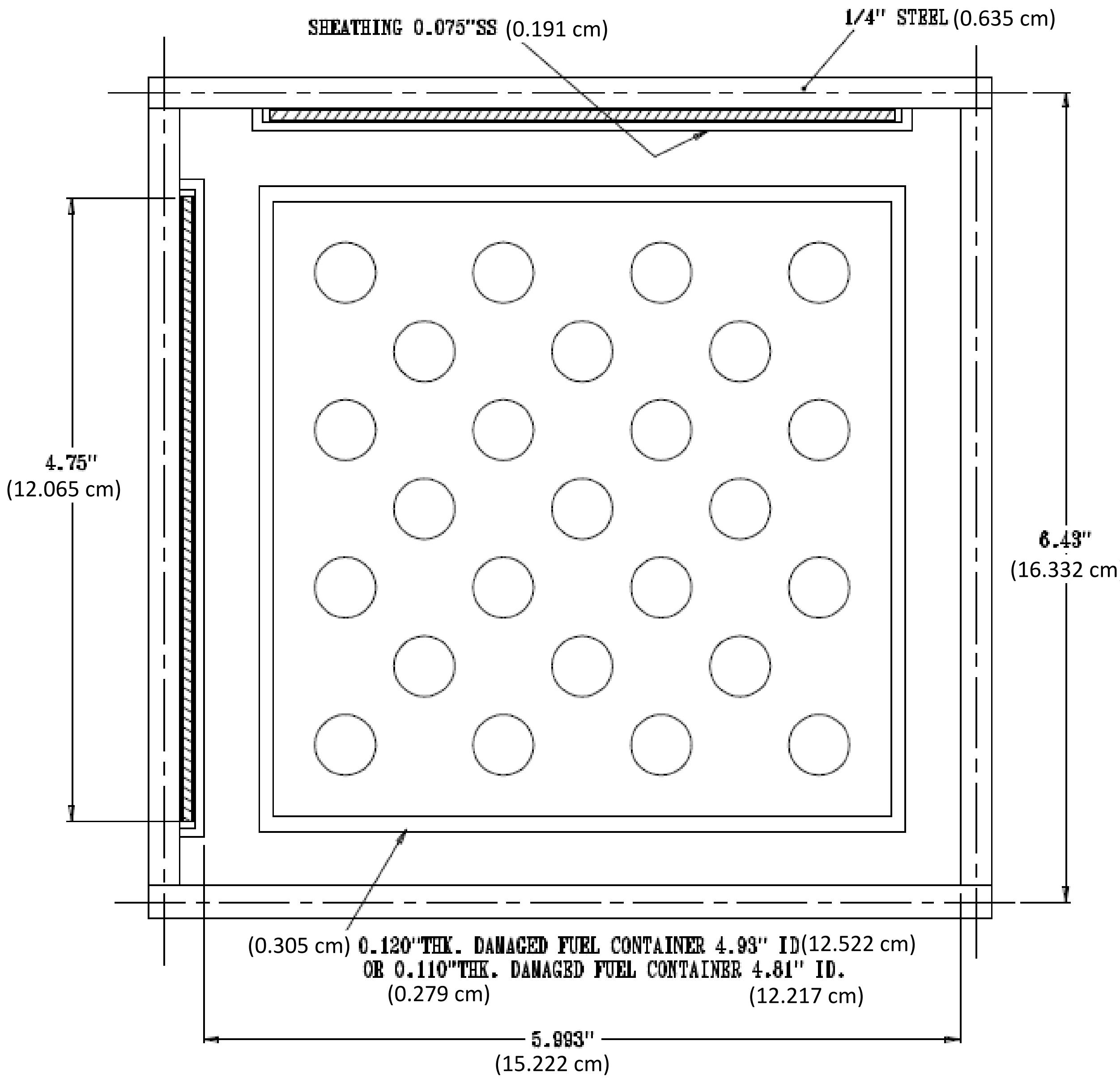

FIGURE 6.4.8 FAILED FUEL CALCUALTION MODEL (PLANAR CROSS-SECTION) WITH 7X7 ARRAY WITH 24 MISSING RODS IN THE MPC-68 BASKET (SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

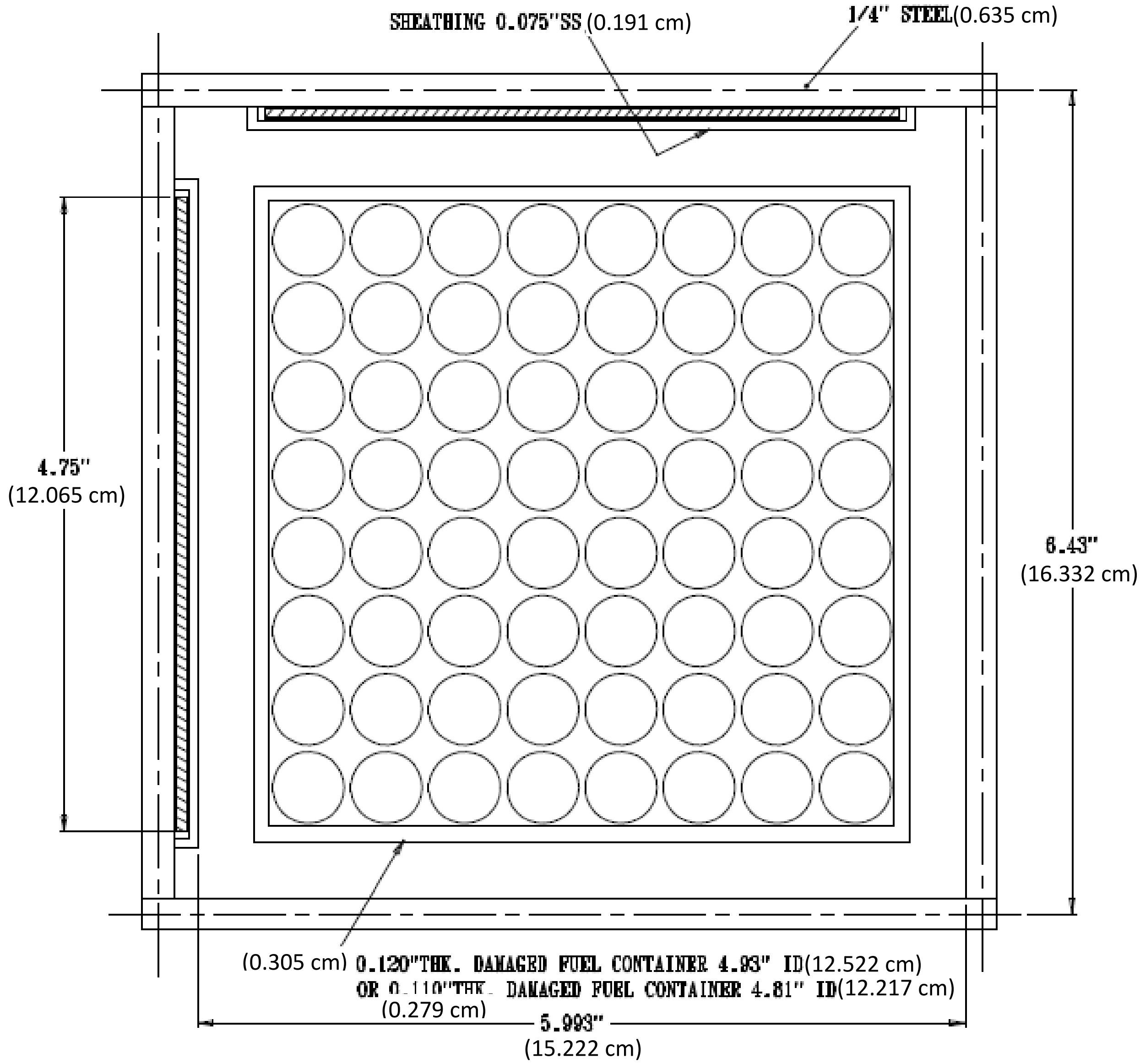

FIGURE 6.4.9 FAILED FUEL CALCUALTION MODEL (PLANAR CROSS-SECTION) WITH DAMAGED FUEL COLLAPSED INTO 8X8 ARRAY IN THE MPC-68 BASKET
(SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

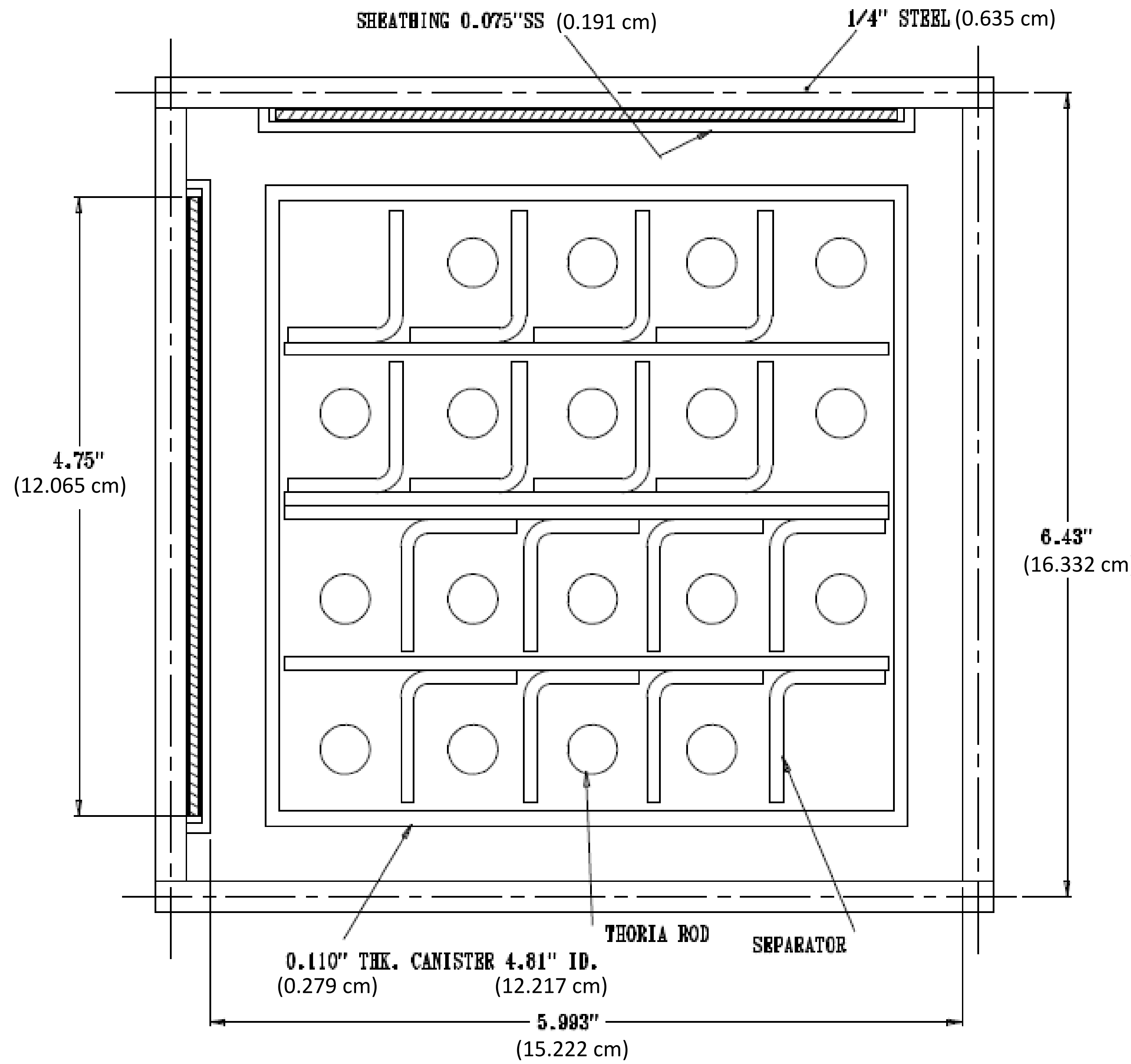

FIGURE 6.4.10 THORIA ROD CANISTER (PLANAR CROSS-SECTION) WITH 18 THORIA RODS IN THE MPC-68 BASKET
(SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

6.5 CRITICALITY BENCHMARK EXPERIMENTS

Benchmark calculations have been made on selected critical experiments, chosen, insofar as possible, to bound the range of variables in the cask designs. The most important parameters are (1) the enrichment, (2) the water-gap size (MPC-24) or cell spacing (*MPC-32 and* MPC-68), (3) the ^{10}B loading of the neutron absorber panels, *and (4) the soluble boron concentration in the* water (MPC-24 and MPC-32*)*. Other parameters, within the normal range of cask and fuel designs, have a smaller effect, but are also included. No significant trends were evident in the benchmark calculations or the derived bias. Detailed benchmark calculations are presented in Appendix 6.A.

The benchmark calculations were performed with the same computer codes and cross-section data, described in Section 6.4, that were used to calculate the k_{eff} values for the cask. Further, all calculations were performed on the same computer hardware, specifically, personal computers using the pentium processor.

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6.6 REGULATORY COMPLIANCE

This chapter documents the criticality evaluation of the HI-STAR 100 System for the storage of spent nuclear fuel. This evaluation demonstrates that the HI-STAR 100 System is in full compliance with the criticality requirements of 10CFR72 and NUREG-1536.

Structures, systems, and components important to criticality safety are described in sufficient detail in this chapter to enable an evaluation of their effectiveness.

The HI-STAR 100 System is designed to be subcritical under all credible conditions. The criticality design is based on favorable geometry and fixed neutron poisons. An appraisal of the fixed neutron poisons has shown that they will remain effective for a storage period greater than 20 years, and there is no credible way to lose it, therefore there is no need to provide a positive means to verify their continued efficacy as required by 10CFR72.124(b).

The criticality evaluation has demonstrated that the cask will enable the storage of spent fuel for a minimum of 20 years with an adequate margin of safety. Further, the evaluation has demonstrated that the design basis accidents have no adverse effect on the design parameters important to criticality safety, and therefore, the HI-STAR 100 System is in full compliance with the double contingency requirements of 10CRF72.124. Therefore, it is concluded that the criticality design features for the HI-STAR 100 System are in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The criticality evaluation provides reasonable assurance that the HI-STAR 100 System will allow safe storage of spent fuel.

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APPENDIX 6.A: BENCHMARK CALCULATIONS

6.A.1 INTRODUCTION AND SUMMARY

Benchmark calculations have been made on selected critical experiments, chosen, in so far as possible, to bound the range of variables in the cask designs. Two independent methods of analysis were used, differing in cross section libraries and in the treatment of the cross sections. MCNP4a [6.A.1] is a continuous energy Monte Carlo code and KENO5a [6.A.2] uses group-dependent cross sections. For the KENO5a analyses reported here, the 238-group library was chosen, processed through the NITAWL-II [6.A.2] program to create a working library and to account for resonance self-shielding in uranium-238 (Nordheim integral treatment). The 238 group library was chosen to avoid or minimize the errors† (trends) that have been reported (e.g., [6.A.3 through 6.A.5]) for calculations with collapsed cross section sets.

In cask designs, the three most significant parameters affecting criticality are (1) the fuel enrichment, (2) the ^{10}B loading in the neutron absorber, and (3) the lattice spacing (or water-gap thickness if a flux-trap design is used). Other parameters, within the normal range of cask and fuel designs, have a smaller effect, but are also included in the analyses.

Table 6.A.1 summarizes results of the benchmark calculations for all cases selected and analyzed, as referenced in the table. The effect of the major variables are discussed in subsequent sections below. It is important to note that there is obviously considerable overlap in parameters since it is not possible to vary a single parameter and maintain criticality; some other parameter or parameters must be concurrently varied to maintain criticality.

One possible way of representing the data is through a spectrum index that incorporates all of the variations in parameters. KENO5a computes and prints the "energy of the average lethargy causing fission". In MCNP4a, by utilizing the tally option with the identical 238-group energy structure as in KENO5a, the number of fissions in each group may be collected and the energy of the average lethargy causing fission determined (post-processing).

Figures 6.A.1 and 6.A.2 show the calculated k_{eff} for the benchmark critical experiments as a function of the "energy of the average lethargy causing fission" for MCNP4a and KENO5a, respectively (UO_2 fuel only). The scatter in the data (even for comparatively minor variation in

† Small but observable trends (errors) have been reported for calculations with the 27-group and 44-group collapsed libraries. These errors are probably due to the use of a single collapsing spectrum when the spectrum should be different for the various cases analyzed, as evidenced by the spectrum indices.

critical parameters) represents experimental error[†] in performing the critical experiments within each laboratory, as well as between the various testing laboratories. The B&W critical experiments show a larger experimental error than the PNL criticals. This would be expected since the B&W criticals encompass a greater range of critical parameters than the PNL criticals.

Linear regression analysis of the data in Figures 6.A.1 and 6.A.2 show that there are no trends, as evidenced by very low values of the correlation coefficient (0.13 for MCNP4a and 0.21 for KENO5a). The total bias (systematic error, or mean of the deviation from a k_{eff} of exactly 1.000) for the two methods of analysis are shown in the table below.

Calculational Bias of MCNP4a and KENO5a		
	Total	Truncated
MCNP4a	0.0009 ± 0.0011	0.0021 ± 0.0006
KENO5a	0.0030 ± 0.0012	0.0036 ± 0.0009

The values of bias shown in this table include both the bias derived directly from the calculated k_{eff} values in Table 6.A.1, and a more conservative value derived by arbitrarily truncating to 1.000 any calculated value that exceeds 1.000. The bias and standard error of the bias were calculated by the following equations[††], with the standard error multiplied by the one-sided K-factor for 95% probability at the 95% confidence level from NBS Handbook 91 [6.A.18] (for the number of cases analyzed, the K-factor is ~2.05 or slightly more than 2).

$$\bar{k} = \frac{1}{n}\sum_{i=1}^{n} k_i \qquad (6.A.1)$$

† A classical example of experimental error is the corrected enrichment in the PNL experiments, first as an addendum to the initial report and, secondly, by revised values in subsequent reports for the same fuel rods.

†† These equations may be found in any standard text on statistics, for example, reference [6.A.6] (or the MCNP4a manual) and is the same methodology used in MCNP4a and in KENO5a.

$$\sigma_{\bar{k}}^2 = \frac{\sum_{i=1}^{n} k_i^2 - (\sum_{i=1}^{n} k_i)^2 / n}{n(n-1)} \qquad (6.A.2)$$

$$\text{Bias} = (1 - \bar{k}) \pm K\sigma_{\bar{k}} \qquad (6.A.3)$$

where k_i are the calculated reactivities for n critical experiments; $\sigma_{\bar{k}}$ is the unbiased estimator of the standard deviation of the mean (also called the standard error of the bias (mean)); and K is the one-sided multiplier for 95% probability at the 95% confidence level (NBS Handbook 91 [6.A.18]).

Formula 6.A.3 is based on the methodology of the National Bureau of Standards (now NIST) and is used to calculate the values presented on page 6.A-2. The first portion of the equation, (1-$\bar{k}$), is the actual bias which is added to the MCNP4a and KENO5a results. The second term, $K\sigma_{\bar{k}}$, which corresponds to σ_B in Section 6.4.3, is the uncertainty or standard error associated with the bias. The K values used were obtained from the National Bureau of Standards Handbook 91 and are for one-sided statistical tolerance limits for 95% probability at the 95% confidence level. The actual K values for the 56 critical experiments evaluated with MCNP4a and the 53 critical experiments evaluated with KENO5a are 2.04 and 2.05, respectively.

The larger of the calculational biases (truncated bias) was used to evaluate the maximum k_{eff} values for the cask designs.

6.A.2 Effect of Enrichment

The benchmark critical experiments include those with enrichments ranging from 2.46% to 5.74% and therefore span the enrichment range for the MPC designs. Figures 6.A.3 and 6.A.4 show the calculated k_{eff} values (Table 6.A.1) as a function of the fuel enrichment reported for the critical experiments. Linear regression analyses for these data confirms that there are no trends, as indicated by low values of the correlation coefficients (0.03 for MCNP4a and 0.38 for KENO5a). Thus, there are no corrections to the bias for the various enrichments.

As further confirmation of the absence of any trends with enrichment, the MPC-68 configuration was calculated with both MCNP4a and KENO5a for various enrichments. The cross-comparison of calculations with codes of comparable sophistication is suggested in Reg. Guide 3.41. Results of this comparison, shown in Table 6.A.2 and Figure 6.A.5, confirm no significant difference in the calculated values of k_{eff} for the two independent codes as evidenced by the 45° slope of the curve. Since it is very unlikely that two independent methods of analysis would be subject to the same error, this comparison is considered confirmation of the absence of an enrichment effect (trend) in the bias.

6.A.3 Effect of ^{10}B Loading

Several laboratories have performed critical experiments with a variety of thin absorber panels similar to the Boral panels in the cask designs. Of these critical experiments, those performed by B&W are the most representative of the cask designs. PNL has also made some measurements with absorber plates, but, with one exception (a flux-trap experiment), the reactivity worth of the absorbers in the PNL tests is very low and any significant errors that might exist in the treatment of strong thin absorbers could not be revealed.

Table 6.A.3 lists the subset of experiments using thin neutron absorbers (from Table 6.A.1) and shows the reactivity worth (Δk) of the absorber.†

No trends with reactivity worth of the absorber are evident, although based on the calculations shown in Table 6.A.3, some of the B&W critical experiments seem to have unusually large experimental errors. B&W made an effort to report some of their experimental errors. Other laboratories did not evaluate their experimental errors.

To further confirm the absence of a significant trend with ^{10}B concentration in the absorber, a cross-comparison was made with MCNP4a and KENO5a (as suggested in Reg. Guide 3.41). Results are shown in Figure 6.A.6 and Table 6.A.4 for the MPC-68 cask†† geometry. These data substantiate the absence of any error (trend) in either of the two codes for the conditions analyzed (data points fall on a 45° line, within an expected 95% probability limit).

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† The reactivity worth of the absorber panels was determined by repeating the calculation with the absorber analytically removed and calculating the incremental (Δk) change in reactivity due to the absorber.

†† The MPC-68 geometry was chosen for this comparison since it contains the greater number of Boral panels and would therefore be expected to be the most sensitive to trends (errors) in calculations.

6.A.4 Miscellaneous and Minor Parameters

6.A.4.1 Reflector Material and Spacings

PNL has performed a number of critical experiments with thick steel and lead reflectors.† Analysis of these critical experiments are listed in Table 6.A.5 (subset of data in Table 6.A.1). There appears to be a small tendency toward overprediction of k_{eff} at the lower spacing, although there are an insufficient number of data points in each series to allow a quantitative determination of any trends. The tendency toward overprediction at close spacing means that the cask calculations may be slightly more conservative than otherwise.

6.A.4.2 Fuel Pellet Diameter and Lattice Pitch

The critical experiments selected for analysis cover a range of fuel pellet diameters from 0.311 to 0.444 inches (0.790 to 1.128 cm), and lattice spacings from 0.476 to 1.00 inches (1.209 to 2.54 cm). In the cask designs, the fuel pellet diameters range from 0.303 to 0.3835 inches (0.770 to 0.974 cm) O.D. (0.496 to 0.580 inch (1.260 to 1.473 cm) lattice spacing) for PWR fuel and from 0.3224 to 0.498 inches (0.819 to 1.265 cm) O.D. (0.488 to 0.740 inch (1.240 to 1.880 cm) lattice spacing) for BWR fuel. Thus, the critical experiments analyzed provide a reasonable representation of the fuel in the MPC designs. Based on the data in Table 6.A.1, there does not appear to be any observable trend with either fuel pellet diameter or lattice pitch, at least over the range of the critical experiments or the cask designs.

6.A.4.3 Soluble Boron Concentration Effects

Various soluble boron concentrations were used in the B&W series of critical experiments and in one PNL experiment, with boron concentrations ranging up to 2550 ppm. Results of MCNP4a (and one KENO5a) calculations are shown in Table 6.A.6. Analyses of the very high boron concentration experiments (>1300 ppm) show a tendency to slightly overpredict reactivity for the three experiments exceeding 1300 ppm. In turn, this would suggest that the evaluation of the MPC-32 with various soluble boron concentration could be slightly conservative for the high soluble boron concentration.

6.A.5 MOX Fuel

The number of critical experiments with PuO_2 bearing fuel (MOX) is more limited than for UO_2 fuel. However, a number of MOX critical experiments have been analyzed and the results are

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†Parallel experiments with a depleted uranium reflector were also performed but not included in the present analysis since they are not pertinent to the Holtec cask design. A lead reflector is also not directly pertinent, but might be used in future designs.

shown in Table 6.A.7. Results of these analyses are generally above a k_{eff} of 1.00, indicating that when Pu is present, MCNP4a and KENO5a overpredict the reactivity.

This may indicate that calculation for MOX fuel will be expected to be conservative, especially with MCNP4a. It may be noted that for the larger lattice spacings, the KENO5a calculated reactivities are below 1.00, suggested that a small trend may exist with KENO5a. It is also possible that the overprediction in k_{eff} in both codes may be due to a small inadequacy in the determination of the Pu-241 decay and Am-241 growth. This possibility is supported by the consistency in calculated k_{eff} over a wide range of the spectral index (energy of the average lethargy causing fission).

6.A.6 References

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Table 6.A.1
Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF (eV) MCNP4a	EALF (eV) KENO5a
1	B&W-1484 (6.A.7)	Core I	2.46	0.9964 ± 0.0010	0.9898 ± 0.0006	0.1759	0.1753
2	B&W-1484 (6.A.7)	Core II	2.46	1.0008 ± 0.0011	1.0015 ± 0.0005	0.2553	0.2446
3	B&W-1484 (6.A.7)	Core III	2.46	1.0010 ± 0.0012	1.0005 ± 0.0005	0.1999	0.1939
4	B&W-1484 (6.A.7)	Core IX	2.46	0.9956 ± 0.0012	0.9901 ± 0.0006	0.1422	0.1426
5	B&W-1484 (6.A.7)	Core X	2.46	0.9980 ± 0.0014	0.9922 ± 0.0006	0.1513	0.1499
6	B&W-1484 (6.A.7)	Core XI	2.46	0.9978 ± 0.0012	1.0005 ± 0.0005	0.2031	0.1947
7	B&W-1484 (6.A.7)	Core XII	2.46	0.9988 ± 0.0011	0.9978 ± 0.0006	0.1718	0.1662
8	B&W-1484 (6.A.7)	Core XIII	2.46	1.0020 ± 0.0010	0.9952 ± 0.0006	0.1988	0.1965
9	B&W-1484 (6.A.7)	Core XIV	2.46	0.9953 ± 0.0011	0.9928 ± 0.0006	0.2022	0.1986
10	B&W-1484 (6.A.7)	Core XV††	2.46	0.9910 ± 0.0011	0.9909 ± 0.0006	0.2092	0.2014
11	B&W-1484 (6.A.7)	Core XVI††	2.46	0.9935 ± 0.0010	0.9889 ± 0.0006	0.1757	0.1713
12	B&W-1484 (6.A.7)	Core XVII	2.46	0.9962 ± 0.0012	0.9942 ± 0.0005	0.2083	0.2021

Table 6.A.1
Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF (eV) MCNP4a	EALF (eV) KENO5a
13	B&W-1484 (6.A.7)	Core XVIII	2.46	1.0036 ± 0.0012	0.9931 ± 0.0006	0.1705	0.1708
14	B&W-1484 (6.A.7)	Core XIX	2.46	0.9961 ± 0.0012	0.9971 ± 0.0005	0.2103	0.2011
15	B&W-1484 (6.A.7)	Core XX	2.46	1.0008 ± 0.0011	0.9932 ± 0.0006	0.1724	0.1701
16	B&W-1484 (6.A.7)	Core XXI	2.46	0.9994 ± 0.0010	0.9918 ± 0.0006	0.1544	0.1536
17	B&W-1645 (6.A.8)	S-type Fuel, w/886 ppm B	2.46	0.9970 ± 0.0010	0.9924 ± 0.0006	1.4475	1.4680
18	B&W-1645 (6.A.8)	S-type Fuel, w/746 ppm B	2.46	0.9990 ± 0.0010	0.9913 ± 0.0006	1.5463	1.5660
19	B&W-1645 (6.A.8)	SO-type Fuel, w/1156 ppm B	2.46	0.9972 ± 0.0009	0.9949 ± 0.0005	0.4241	0.4331
20	B&W-1810 (6.A.9)	Case 1 1337 ppm B	2.46	1.0023 ± 0.0010	NC	0.1531	NC
21	B&W-1810 (6.A.9)	Case 12 1899 ppm B	2.46/4.02	1.0060 ± 0.0009	NC	0.4493	NC
22	French (6.A.10)	Water Moderator 0 gap	4.75	0.9966 ± 0.0013	NC	0.2172	NC
23	French (6.A.10)	Water Moderator 2.5 cm gap	4.75	0.9952 ± 0.0012	NC	0.1778	NC
24	French (6.A.10)	Water Moderator 5 cm gap	4.75	0.9943 ± 0.0010	NC	0.1677	NC

Table 6.A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF (eV) MCNP4a	EALF (eV) KENO5a
25	French (6.A.10)	Water Moderator 10 cm gap	4.75	0.9979 ± 0.0010	NC	0.1736	NC
26	PNL-3602 (6.A.11)	Steel Reflector, 0 cm separation	2.35	NC	1.0004 ± 0.0006	NC	0.1018
27	PNL-3602 (6.A.11)	Steel Reflector, 1.321 cm separation	2.35	0.9980 ± 0.0009	0.9992 ± 0.0006	0.1000	0.0909
28	PNL-3602 (6.A.11)	Steel Reflector, 2.616 cm separation	2.35	0.9968 ± 0.0009	0.9964 ± 0.0006	0.0981	0.0975
29	PNL-3602 (6.A.11)	Steel Reflector, 3.912 cm separation	2.35	0.9974 ± 0.0010	0.9980 ± 0.0006	0.0976	0.0970
30	PNL-3602 (6.A.11)	Steel Reflector, Infinite separation	2.35	0.9962 ± 0.0008	0.9939 ± 0.0006	0.0973	0.0968
31	PNL-3602 (6.A.11)	Steel Reflector, 0 cm separation	4.306	NC	1.0003 ± 0.0007	NC	0.3282
32	PNL-3602 (6.A.11)	Steel Reflector, 1.321 cm separation	4.306	0.9997 ± 0.0010	1.0012 ± 0.0007	0.3016	0.3039
33	PNL-3602 (6.A.11)	Steel Reflector, 2.616 cm separation	4.306	0.9994 ± 0.0012	0.9974 ± 0.0007	0.2911	0.2927
34	PNL-3602 (6.A.11)	Steel Reflector, 5.405 cm separation	4.306	0.9969 ± 0.0011	0.9951 ± 0.0007	0.2828	0.2860
35	PNL-3602 (6.A.11)	Steel Reflector, Infinite separation	4.306	0.9910 ± 0.0020	0.9947 ± 0.0007	0.2851	0.2864
36	PNL-3602 (6.A.11)	Steel Reflector, with Boral Sheets	4.306	0.9941 ± 0.0011	0.9970 ± 0.0007	0.3135	0.3150

Table 6.A.1
Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF (eV) MCNP4a	EALF (eV) KENO5a
37	PNL-3626 (6.A.12)	Lead Reflector, 0 cm sepn.	4.306	NC	1.0003 ± 0.0007	NC	0.3159
38	PNL-3626 (6.A.12)	Lead Reflector, 0.55 cm sepn.	4.306	1.0025 ± 0.0011	0.9997 ± 0.0007	0.3030	0.3044
39	PNL-3626 (6.A.12)	Lead Reflector, 1.956 cm sepn.	4.306	1.0000 ± 0.0012	0.9985 ± 0.0007	0.2883	0.2930
40	PNL-3626 (6.A.12)	Lead Reflector, 5.405 cm sepn.	4.306	0.9971 ± 0.0012	0.9946 ± 0.0007	0.2831	0.2854
41	PNL-2615 (6.A.13)	Experiment 004/032 – no absorber	4.306	0.9925 ± 0.0012	0.9950 ± 0.0007	0.1155	0.1159
42	PNL-2615 (6.A.13)	Experiment 030 – Zr plates	4.306	NC	0.9971 ± 0.0007	NC	0.1154
43	PNL-2615 (6.A.13)	Experiment 013 – Steel plates	4.306	NC	0.9965 ± 0.0007	NC	0.1164
44	PNL-2615 (6.A.13)	Experiment 014 – Steel plates	4.306	NC	0.9972 ± 0.0007	NC	0.1164
45	PNL-2615 (6.A.13)	Exp. 009 1.05% Boron Steel plates	4.306	0.9982 ± 0.0010	0.9981 ± 0.0007	0.1172	0.1162
46	PNL-2615 (6.A.13)	Exp. 009 1.62% Boron Steel plates	4.306	0.9996 ± 0.0012	0.9982 ± 0.0007	0.1161	0.1173
47	PNL-2615 (6.A.13)	Exp. 031 – Boral plates	4.306	0.9994 ± 0.0012	0.9969 ± 0.0007	0.1165	0.1171
48	PNL-7167 (6.A.14)	Experiment 214R – with flux traps	4.306	0.9991 ± 0.0011	0.9956 ± 0.0007	0.3722	0.3812

Table 6.A.1
Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF (eV) MCNP4a	EALF (eV) KENO5a
49	PNL-7167 (6.A.14)	Experiment 214V3 –with flux trap	4.306	0.9969 ± 0.0011	0.9963 ± 0.0007	0.3742	0.3826
50	PNL-4267 (6.A.15)	Case 173 – 0 ppm B	4.306	0.9974 ± 0.0012	NC	0.2893	NC
51	PNL-4267 (6.A.15)	Case 177 – 2550 ppm B	4.306	1.0057 ± 0.0010	NC	0.5509	NC
52	PNL-5803 (6.A.16)	MOX Fuel – Type 3.2 Exp. 21	20% Pu	1.0041 ± 0.0011	1.0046 ± 0.0006	0.9171	0.8868
53	PNL-5803 (6.A.16)	MOX Fuel – Type 3.2 Exp. 43	20% Pu	1.0058 ± 0.0012	1.0036 ± 0.0006	0.2968	0.2944
54	PNL-5803 (6.A.16)	MOX Fuel – Type 3.2 Exp. 13	20% Pu	1.0083 ± 0.0011	0.9989 ± 0.0006	0.1665	0.1706
55	PNL-5803 (6.A.16)	MOX Fuel – Type 3.2 Exp. 32	20% Pu	1.0079 ± 0.0011	0.9966 ± 0.0006	0.1339	0.1165
56	WCAP-3385 (6.A.17)	Saxton Case 52 PuO_2 0.52” pitch	6.6% Pu	0.9996 ± 0.0011	1.0005 ± 0.0006	0.8665	0.8417
57	WCAP-3385 (6.A.17)	Saxton Case 52 U 0.52” pitch	5.74	1.0000 ± 0.0010	0.9956 ± 0.0007	0.4476	0.4580
58	WCAP-3385 (6.A.17)	Saxton Case 56 PuO_2 0.56” pitch	6.6% Pu	1.0036 ± 0.0011	1.0047 ± 0.0006	0.5289	0.5197
59	WCAP-3385 (6.A.17)	Saxton Case 56 borated PuO_2	6.6% Pu	1.0008 ± 0.0010	NC	0.6389	NC
60	WCAP-3385 (6.A.17)	Saxton Case 56 U 0.56” pitch	5.74	0.9994 ± 0.0011	0.9967 ± 0.0007	0.2923	0.2954

Table 6.A.1
Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF (eV) MCNP4a	EALF (eV) KENO5a
61	WCAP-3385 (6.A.17)	Saxton Case 79 PuO_2 0.79” pitch	6.6% Pu	1.0063 ± 0.0011	1.0133 ± 0.0006	0.1520	0.1555
62	WCAP-3385 (6.A.17)	Saxton Case 79 U 0.79” pitch	5.74	1.0039 ± 0.0011	1.0008 ± 0.0006	0.1036	0.1047

Notes: NC stands for not calculated.

† EALF is the energy of the average lethargy causing fission

†† The experimental results appear to be statistical outliers (>3σ) suggesting the possibility of unusually large experimental error. Although they could be justifiably excluded, for conservatism, they were retained in determining the calculational basis.

Table 6.A.2

COMPARISON OF MCNP4a AND KENO5a CALCULATED REACTIVITIES[†]
FOR VARIOUS ENRICHMENTS (UO_2)

Enrichment	Calculated $k_{eff} \pm 1\sigma$	
	MCNP4a	**KENO5a**
3.0	0.8465 ± 0.0011	0.8478 ± 0.0004
3.5	0.8820 ± 0.0011	0.8841 ±0.0004
3.75	0.9019 ± 0.0011	0.8987 ± 0.0004
4.0	0.9132 ± 0.0010	0.9140 ± 0.0004
4.2	0.9276 ± 0.0011	0.9237 ± 0.0004
4.5	0.9400 ± 0.0011	0.9388 ± 0.0004

[†] Based on the MPC-68 with the GE 8x8R

Table 6.A.3

MCNP4a CALCULATED REACTIVITIES FOR
CRITICAL EXPERIMENTS WITH NEUTRON ABSORBERS (UO_2)

Ref.	Experiment		Δk Worth of Absorber	MCNP4a Calculated k_{eff}	EALF† (eV)
6.A.13	PNL-2615	Boral Sheet	0.0139	0.9994 ± 0.0012	0.1165
6.A.7	BAW-1484	Core XX	0.0165	1.0008 ± 0.0011	0.1724
6.A.13	PNL-2615	1.62% Boron-steel	0.0165	0.9996 ± 0.0012	0.1161
6.A.7	BAW-1484	Core XIX	0.0202	0.9961 ± 0.0012	0.2103
6.A.7	BAW-1484	Core XXI	0.0243	0.9994 ± 0.0010	0.1544
6.A.7	BAW-1484	Core XVII	0.0519	0.9962 ± 0.0012	0.2083
6.A.11	PNL-3602	Boral Sheet	0.0708	0.9941 ± 0.0011	0.3135
6.A.7	BAW-1484	Core XV	0.0786	0.9910 ± 0.0011	0.2092
6.A.7	BAW-1484	Core XVI	0.0845	0.9935 ± 0.0010	0.1757
6.A.7	BAW-1484	Core XIV	0.1575	0.9953 ± 0.0011	0.2022
6.A.7	BAW-1484	Core XIII	0.1738	1.0020 ± 0.0011	0.1988
6.A.14	PNL-7167	Expt 214R flux trap	0.1931	0.9991 ± 0.0011	0.3722

† EALF is the energy of the average lethargy causing fission

Table 6.A.4
COMPARISON OF MCNP4a AND KENO5a
CALCULATED REACTIVITIES† FOR VARIOUS BORON LOADINGS (UO_2)

^{10}B, g/cm^2	**Calcualted k_{eff} ± 1σ**	
	MCNP4a	**KENO5a**
0.005	1.0381 ± 0.0012	1.0340 ± 0.0004
0.010	0.9960 ±0.0010	0.9941 ± 0.0004
0.015	0.9727 ± 0.0009	0.9713 ± 0.0004
0.020	0.9541 ± 0.0012	0.9560 ± 0.0004
0.025	0.9433 ± 0.0011	0.9428 ± 0.0004
0.03	0.9325 ± 0.0011	0.9338 ± 0.0004
0.035	0.9234 ± 0.0011	0.9251 ± 0.0004
0.04	0.9173 ± 0.0011	0.9179 ± 0.0004

† based on 4.5% enrichment GE 8x8R in the MPC-68 cask.

Table 6.A.5

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH
THICK LEAD AND STEEL REFLECTORS† (UO_2)

Ref.	Case	Enrichment, wt%	Separation, cm	MCNP4a k_{eff}	KENO5a k_{eff}
6.A.11	Steel Reflector	2.35	1.321	0.9980 ± 0.0009	0.9992 ± 0.0006
		2.35	2.616	0.9968 ± 0.0009	0.9964 ± 0.0006
		2.35	3.912	0.9974 ± 0.0010	0.9980 ± 0.0006
		2.35	∞	0.9962 ± 0.0008	0.9939 ± 0.0006
6.A.11	Steel Reflector	4.306	1.321	0.9997 ± 0.0010	1.0012 ± 0.0007
		4.306	2.616	0.9994 ± 0.0012	0.9974 ± 0.0007
		4.306	3.405	0.9969 ± 0.0011	0.9951 ± 0.0007
		4.306	∞	0.9910 ± 0.0020	0.9947 ± 0.0007
6.A.11	Lead Reflector	4.306	0.55	1.0025 ± 0.0011	0.9997 ± 0.0007
		4.306	1.956	1.0000 ± 0.0012	0.9985 ± 0.0007
		4.306	5.405	0.9971 ± 0.0012	0.9946 ± 0.0007

† Arranged in order of increasing reflector fuel spacing.

Table 6.A.6

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH VARIOUS SOLUBLE BORON CONCENTRATIONS (UO_2)

Reference	**Experiment**	**Boron Concentration ppm**	**Calculated k_{eff}**	
			MCNP4a	**KENO5a**
6.A.15	PNL-4267	0	0.9974 ± 0.0012	-
6.A.8	BAW-1645-4	886	0.9970 ± 0.0010	0.9924 ± 0.0006
6.A.9	BAW-1810	1337	1.0023 ± 0.0010	-
6.A.9	BAW-1810	1899	1.0060 ± 0.0009	-
6.A.15	PNL-4267	2550	1.0057 ± 0.0010	-

Table 6.A.7

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH MOX FUEL

Reference	Case†	MCNP4a		KENO 5a	
		k_{eff}	EALF†† (eV)	k_{eff}	EALF†† (eV)
PNL-5803 [6.A.16]	MOX Fuel – Exp No 21	1.0041±0.0011	0.9171	1.0046±0.0006	0.8868
	MOX Fuel – Exp No 43	1.0058±0.0012	0.2968	1.0036±0.0006	0.2944
	MOX Fuel – Exp No 13	1.0083±0.0011	0.1665	0.9989±0.0006	0.1706
	MOX Fuel – Exp No 32	1.0079±0.0011	0.1139	0.9966±0.0006	0.1165
WCAP- 3385-54 [6.A.17]	Saxton @ 0.52” pitch	0.9996±0.0011	0.8665	1.0005±0.0006	0.8417
	Saxton @ 0.56” pitch	1.0036±0.0011	0.5289	1.0047±0.0006	0.5197
	Saxton @ 0.56” pitch borated	1.0008±0.0010	0.6389	NC	NC
	Saxton @ 0.79” pitch	1.0063±0.0011	0.1520	1.0133±0.0006	0.1555

† Arranged in order of increasing lattice spacing.

†† EALF is the energy of the average lethargy causing fission.

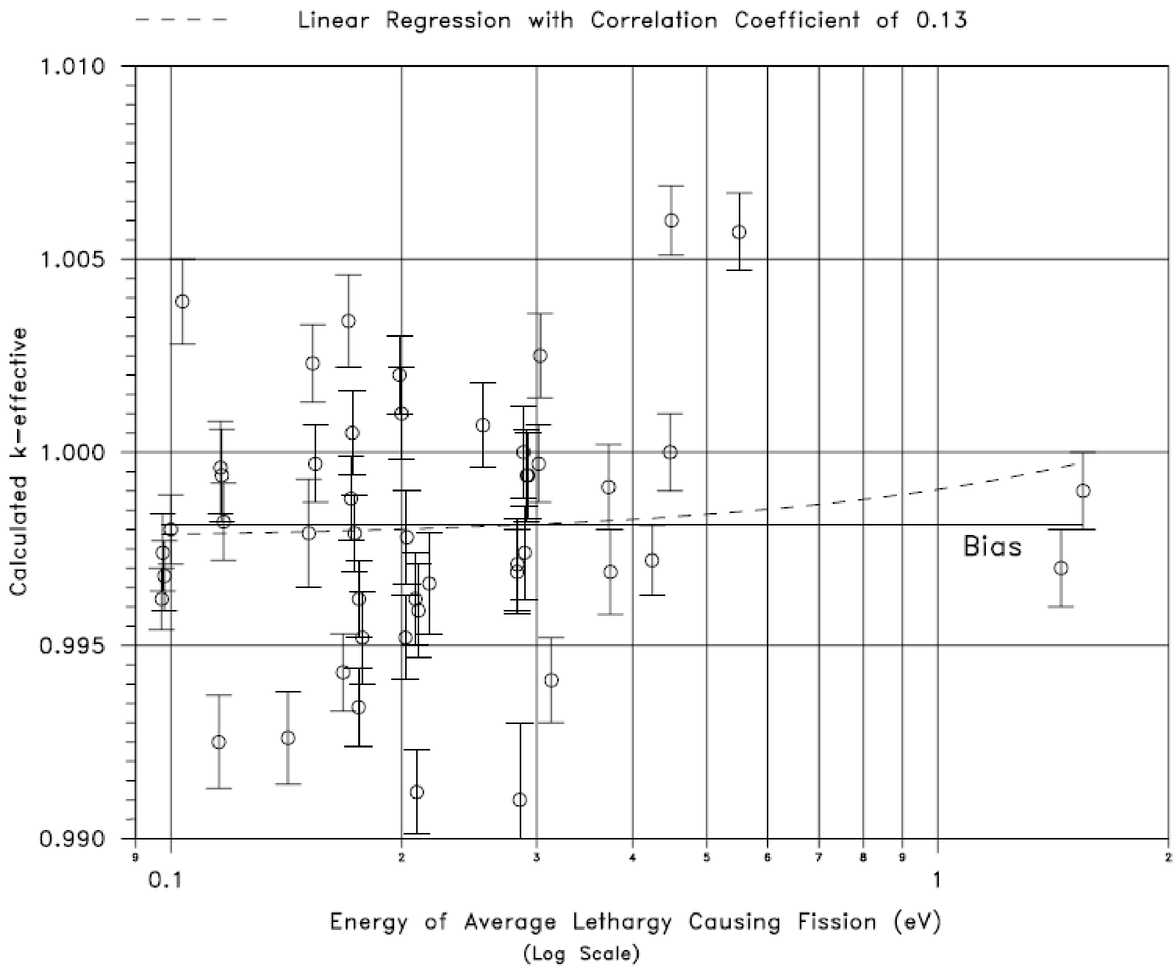

FIGURE 6.A.1 MCNP4a CALCULATED k-eff VALUES FOR VARIOUS VALUES OF THE SPECTRAL INDEX

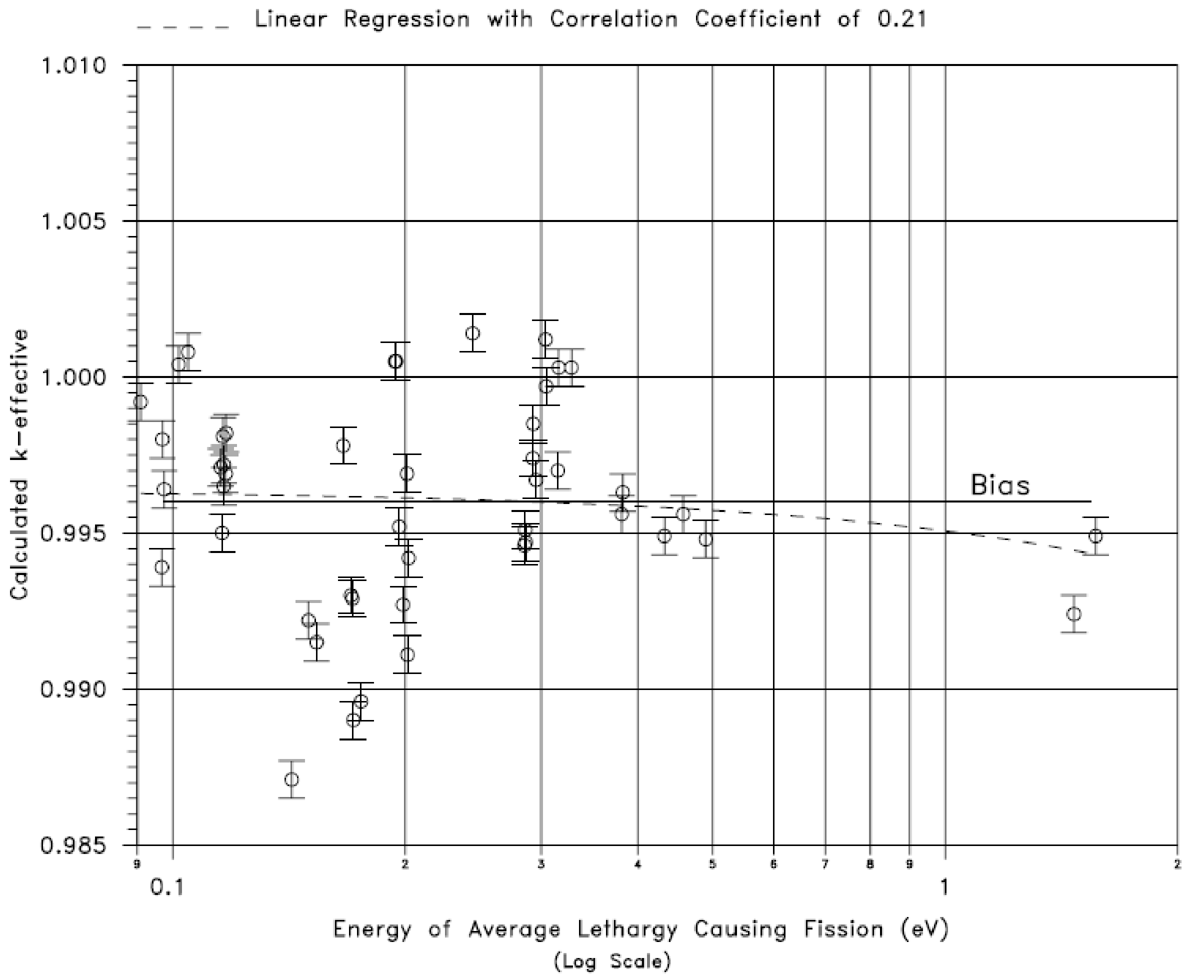

FIGURE 6.A.2 KENO5a CALCULATED k-eff VALUES FOR VARIOUS VALUES OF THE SPECTRAL INDEX

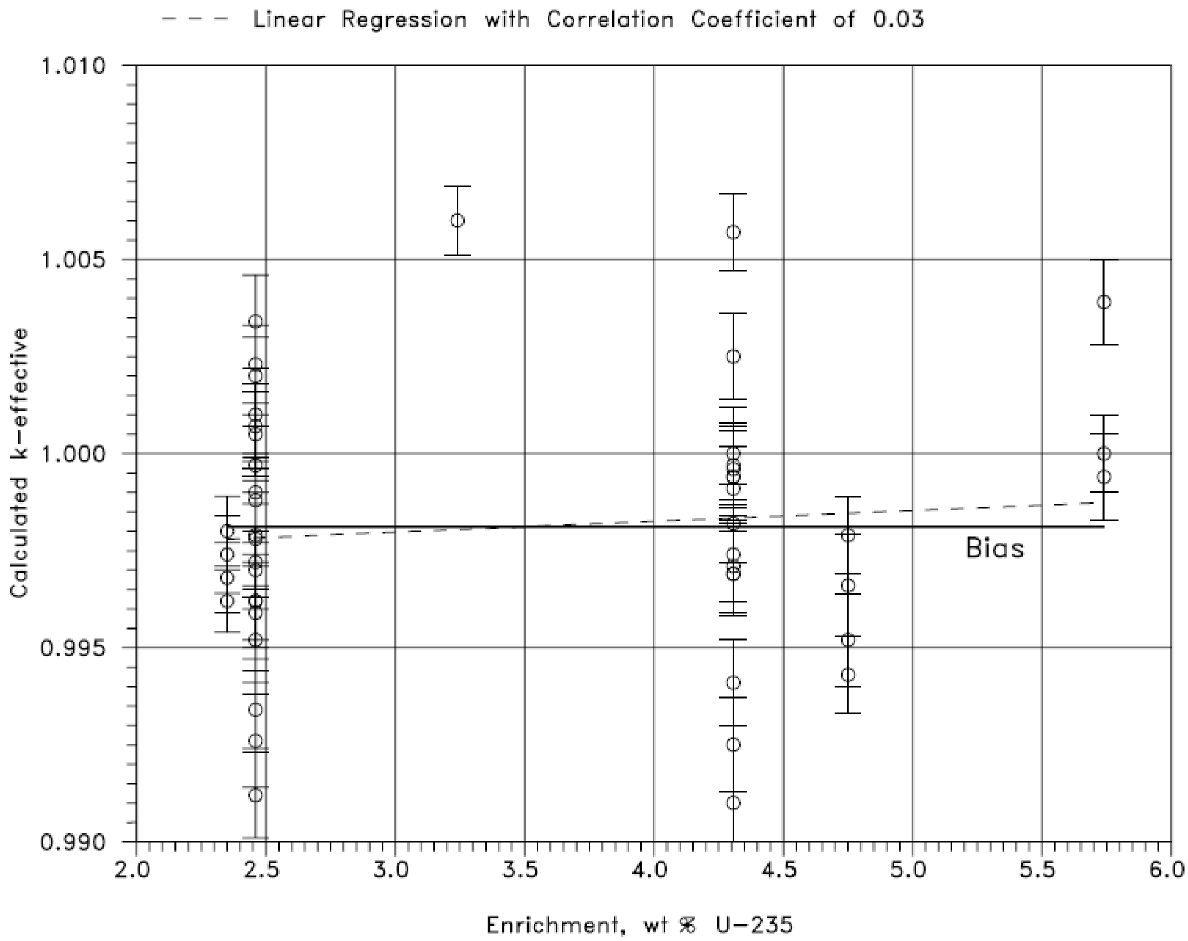

FIGURE 6.A.3 MCNP4a CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

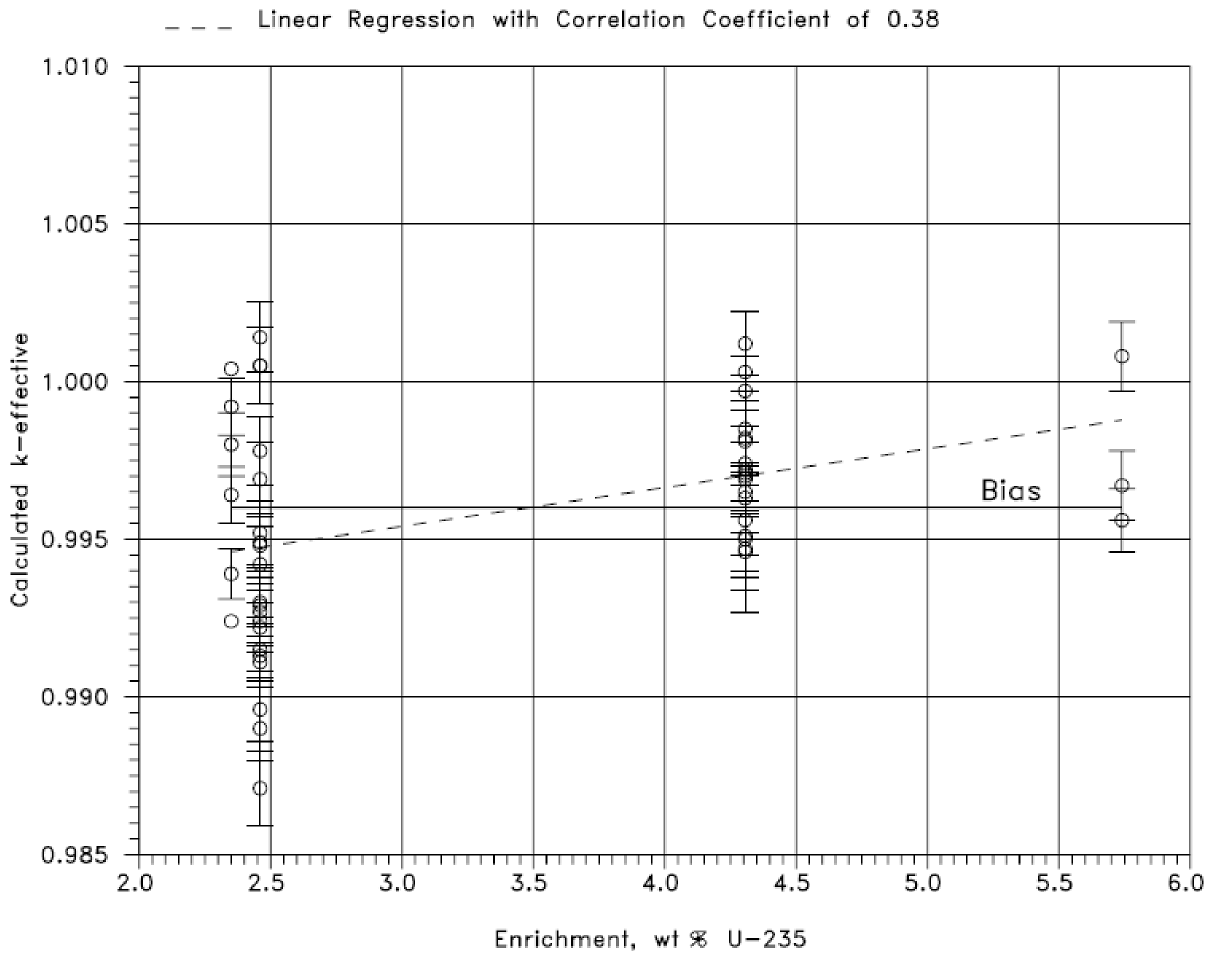

FIGURE 6.A.4 KENO5a CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

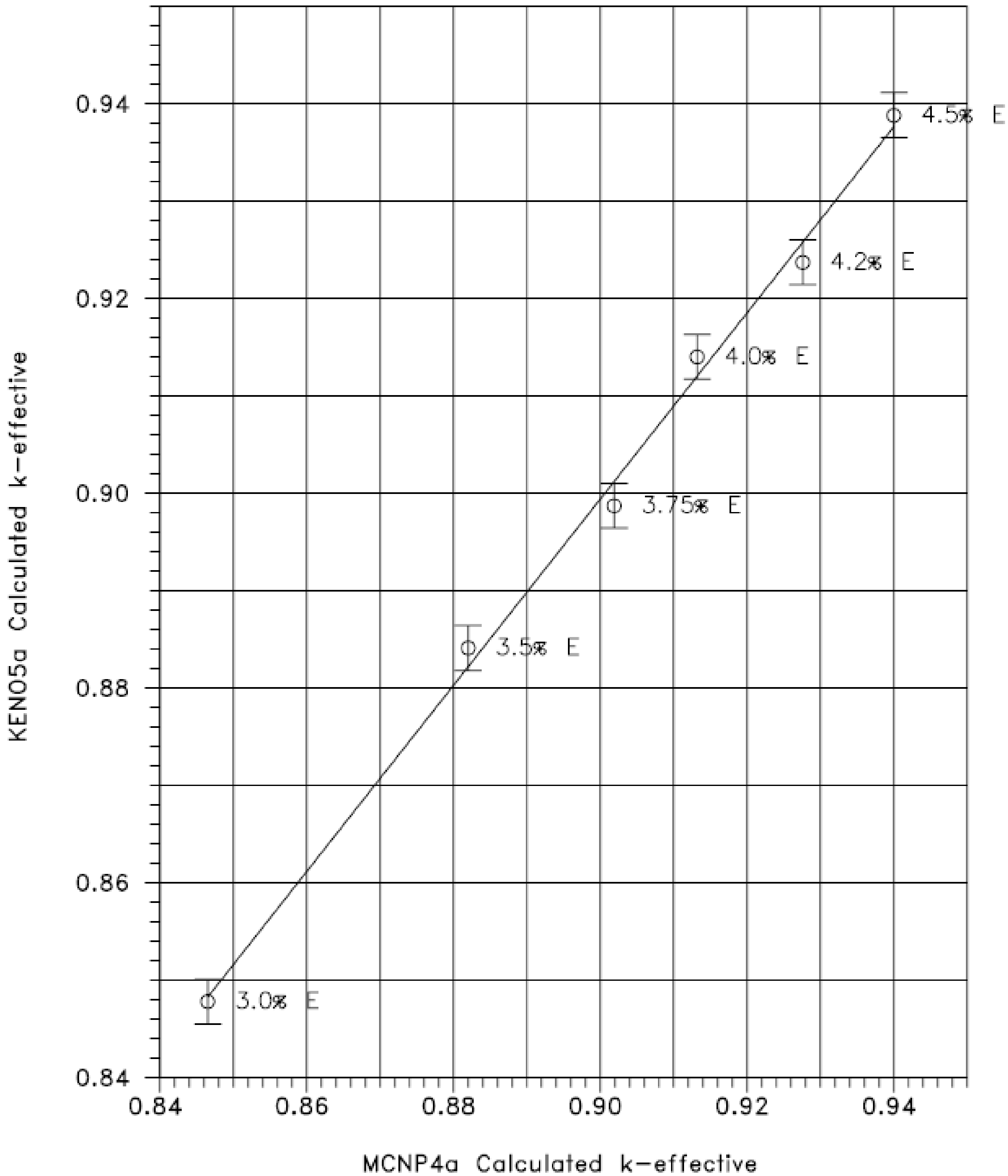

FIGURE 6.A.5 COMPARISON OF MCNP4a AND KENO5a CALCULATIONS FOR VARIOUS FUEL ENRICHMENTS

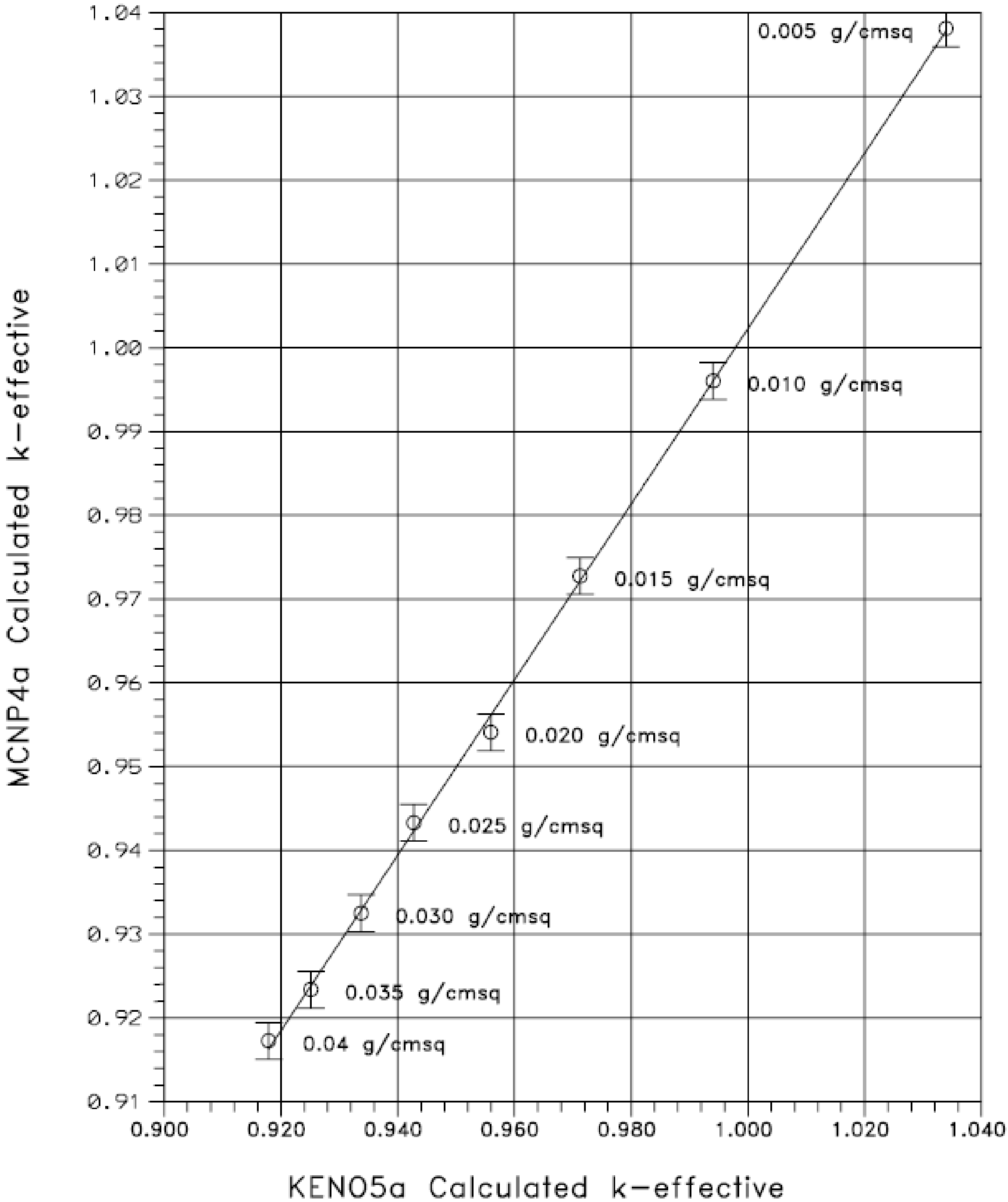

FIGURE 6.A.6 COMPARISON OF MCNP4a AND KENO5a CALCULATIONS FOR VARIOUS BORON-10 AREAL DENSITIES

APPENDIX 6.B: DISTRIBUTED ENRICHMENTS IN BWR FUEL

Fuel assemblies used in BWRs utilize fuel rods of varying enrichments as a means of controlling power peaking during in-core operation. For calculations involving BWR assemblies, the use of a uniform (planar-average) enrichment, as opposed to the distributed enrichments normally used in BWR fuel, produces conservative results. Calculations have been performed to confirm that this statement remains valid in the geometry of the MPC-68. These calculations are based on fuel assembly designs currently in use and two hypothetical distributions, all intended to illustrate that calculations with uniform average enrichments are conservative.

The average enrichment is calculated as the linear average of the various fuel rod enrichments, i.e.,

$$\overline{E} = \frac{1}{n} \sum_{i=1}^{n} E_i ,$$

where E_i is the enrichment in each of the n rods, and $\overline{E}$ is the assembly average enrichment. This parameter conservatively characterizes the fuel assembly and is readily available for specific fuel assemblies in determining the acceptability of the assembly for placement in the MPC-68 cask.

The criticality calculations for average and distributed enrichment cases are compared in Table 6.B.1 to illustrate and confirm the conservatism inherent in using average enrichments. With two exceptions, the cases analyzed represent realistic designs currently in use and encompass fuel with different ratios of maximum pin enrichment to average assembly enrichment. The two exceptions are hypothetical cases intended to extend the models to higher enrichments and to demonstrate that using the average enrichment remains conservative.

Table 6.B.1 shows that, in all cases, the averaged enrichment yields conservative values of reactivity relative to distributed enrichments for both the actual fuel designs and the hypothetical higher enrichment cases. Thus, it is concluded that uniform average enrichments will always yield higher (more conservative) values for reactivity than the corresponding distributed enrichments†.

† This conclusion implicitly assumes the higher enrichment fuel rods are located internal to the assembly (as in BWR fuel), and the lower enriched rods are on the outside.

Table 6.B.1

COMPARISON CALCULATIONS FOR BWR FUEL WITH AVERAGE AND DISTRIBUTED ENRICHMENTS

Case	Average %E	Peak Rod E%	Calculated k_{eff}	
			Average E	**Distributed E**
8x8C04	3.01	3.80	0.8549	0.8429
8x8C04	3.934	4.9	0.9128	0.9029
8x8D05	3.42	3.95	0.8790	0.8708
8x8D05	3.78	4.40	0.9030	0.8974
8x8D05	3.90	4.90	0.9062	0.9042
9x9B01	4.34	4.71	0.9347	0.9285
9x9D01	3.35	4.34	0.8793	0.8583
Hypothetical #1 (48 outer rods of 3.967%E,14 inner rods of 5.0%)	4.20	5.00	0.9289	0.9151
Hypothetical #2 (48 outer rods of 4.354%E,14 inner rods of 5.0%)	4.50	5.00	0.9422	0.9384

APPENDIX 6.C: CALCULATIONAL SUMMARY

The following table lists the maximum k_{eff} (including bias, uncertainties, and calculational statistics), MCNP calculated k_{eff}, standard deviation, and energy of average lethargy causing fission (EALF) for each of the candidate fuel types and basket configurations.

Table 6.C.1
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-24				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A01	0.9295	0.9252	0.0008	0.2084
14x14A02	0.9286	0.9242	0.0008	0.2096
14x14A03	0.9296	0.9253	0.0008	0.2093
14x14B01	0.9159	0.9117	0.0007	0.2727
14x14B02	0.9169	0.9126	0.0008	0.2345
14x14B03	0.9110	0.9065	0.0009	0.2545
14x14B04	0.9084	0.9039	0.0009	0.2563
B14x14B01	0.9228	0.9185	0.0008	0.2675
14x14C01	0.9258	0.9215	0.0008	0.2729
14x14C02	0.9265	0.9222	0.0008	0.2765
14x14C03	0.9287	0.9242	0.0009	0.2825
14x14D01	0.8507	0.8464	0.0008	0.3308
15x15A01	0.9204	0.9159	0.0009	0.2608
15x15B01	0.9369	0.9326	0.0008	0.2632
15C15B02	0.9338	0.9295	0.0008	0.2640
15x15B03	0.9362	0.9318	0.0008	0.2632
15x15B04	0.9370	0.9327	0.0008	0.2612
15x15B05	0.9356	0.9313	0.0008	0.2606
15x15B06	0.9366	0.9324	0.0007	0.2638
B15x15B01	0.9388	0.9343	0.0009	0.2626
15x15C01	0.9255	0.9213	0.0007	0.2493

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-24				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
15x15C02	0.9297	0.9255	0.0007	0.2457
15x15C03	0.9297	0.9255	0.0007	0.2440
15x15C04	0.9311	0.9268	0.0008	0.2435
B15x15C01	0.9361	0.9316	0.0009	0.2385
15x15D01	0.9341	0.9298	0.0008	0.2822
15x15D02	0.9367	0.9324	0.0008	0.2802
15x15D03	0.9354	0.9311	0.0008	0.2844
15x15D04	0.9339	0.9292	0.0010	0.2958
15x15E01	0.9368	0.9325	0.0008	0.2826
15x15F01	0.9395	0.9350	0.0009	0.2903
15x15G01	0.8876	0.8833	0.0008	0.3357
15x15H01	0.9337	0.9292	0.0009	0.2349
16x16A01	0.9287	0.9244	0.0008	0.2704
16x16A02	0.9263	0.9221	0.0007	0.2702
17x17A01	0.9368	0.9325	0.0008	0.2131
17x17A02	0.9368	0.9325	0.0008	0.2131
17x17A03	0.9329	0.9286	0.0008	0.2018
17x17B01	0.9288	0.9243	0.0009	0.2607
17x17B02	0.9290	0.9247	0.0008	0.2596
17x17B03	0.9243	0.9199	0.0008	0.2625
17x17B04	0.9324	0.9279	0.0009	0.2576
17x17B05	0.9266	0.9222	0.0008	0.2539

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-24				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
17x17B06	0.9311	0.9268	0.0008	0.2593
17x17C01	0.9293	0.9250	0.0008	0.2595
17x17C02	0.9336	0.9293	0.0008	0.2624

MPC-68				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
6x6A01	0.7539	0.7498	0.0007	0.2754
6x6A02	0.7517	0.7476	0.0007	0.2510
6x6A03	0.7545	0.7501	0.0008	0.2494
6x6A04	0.7537	0.7494	0.0008	0.2494
6x6A05	0.7555	0.7512	0.0008	0.2470
6x6A06	0.7618	0.7576	0.0008	0.2298
6x6A07	0.7588	0.7550	0.0005	0.2360
6x6A08	0.7808	0.7766	0.0007	0.2527
B6x6A01	0.7888	0.7846	0.0007	0.2310
6x6B01	0.7604	0.7563	0.0007	0.2461
6x6B02	0.7618	0.7577	0.0006	0.2450
6x6B03	0.7619	0.7578	0.0007	0.2439
6x6B04	0.7686	0.7644	0.0008	0.2286
6x6B05	0.7824	0.7785	0.0006	0.2184
B6x6B01	0.7822	0.7783	0.0006	0.2190

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
6x6C01	0.8021	0.7980	0.0007	0.2139
7x7A01	0.7973	0.7930	0.0008	0.2015
7x7B01	0.9372	0.9330	0.0007	0.3658
7x7B02	0.9301	0.9260	0.0007	0.3524
7x7B03	0.9313	0.9271	0.0008	0.3438
7x7B04	0.9311	0.9270	0.0007	0.3816
7x7B05	0.9350	0.9306	0.0008	0.3382
7x7B06	0.9298	0.9260	0.0006	0.3957
B7x7B01	0.9375	0.9332	0.0008	0.3887
B7x7B02	0.9386	0.9344	0.0007	0.3983
8x8A01	0.7685	0.7644	0.0007	0.2227
8x8A02	0.7697	0.7656	0.0007	0.2158
8x8B01	0.9310	0.9265	0.0009	0.2935
8x8B02	0.9227	0.9185	0.0007	0.2993
8x8B03	0.9299	0.9257	0.0008	0.3319
8x8B04	0.9236	0.9194	0.0008	0.3700
B8x8B01	0.9346	0.9301	0.0009	0.3389
B8x8B02	0.9385	0.9343	0.0008	0.3329
B8x8B03	0.9416	0.9375	0.0007	0.3293
8x8C01	0.9315	0.9273	0.0007	0.2822
8x8C02	0.9313	0.9268	0.0009	0.2716
8x8C03	0.9329	0.9286	0.0008	0.2877

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
8x8C04	0.9348	0.9307	0.0007	0.2915
8x8C05	0.9353	0.9312	0.0007	0.2971
8x8C06	0.9353	0.9312	0.0007	0.2944
8x8C07	0.9314	0.9273	0.0007	0.2972
8x8C08	0.9339	0.9298	0.0007	0.2915
8x8C09	0.9301	0.9260	0.0007	0.3183
8x8C10	0.9317	0.9275	0.0008	0.3018
8x8C11	0.9328	0.9287	0.0007	0.3001
8x8C12	0.9285	0.9242	0.0008	0.3062
B8x8C01	0.9357	0.9313	0.0009	0.3141
B8x8C02	0.9425	0.9384	0.0007	0.3081
B8x8C03	0.9418	0.9375	0.0008	0.3056
8x8D01	0.9342	0.9302	0.0006	0.2733
8x8D02	0.9325	0.9284	0.0007	0.2750
8x8D03	0.9351	0.9309	0.0008	0.2731
8x8D04	0.9338	0.9296	0.0007	0.2727
8x8D05	0.9339	0.9294	0.0009	0.2700
8x8D06	0.9365	0.9324	0.0007	0.2777
8x8D07	0.9341	0.9297	0.0009	0.2694
8x8D08	0.9376	0.9332	0.0009	0.2841
B8x8D01	0.9403	0.9363	0.0007	0.2778
8x8E01	0.9312	0.9270	0.0008	0.2831

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
8x8F01	0.9153	0.9111	0.0007	0.2143
9x9A01	0.9353	0.9310	0.0008	0.2875
9x9A02	0.9388	0.9345	0.0008	0.2228
9x9A03	0.9351	0.9310	0.0007	0.2837
9x9A04	0.9396	0.9355	0.0007	0.2262
B9x9A01	0.9417	0.9374	0.0008	0.2236
9x9B01	0.9380	0.9336	0.0008	0.2576
9x9B02	0.9373	0.9329	0.0009	0.2578
9x9B03	0.9417	0.9374	0.0008	0.2545
B9x9B01	0.9436	0.9394	0.0008	0.2506
9x9C01	0.9395	0.9352	0.0008	0.2698
9x9D01	0.9394	0.9350	0.0009	0.2625
9x9E01	0.9402	0.9359	0.0008	0.2249
9x9E02	0.9424	0.9380	0.0008	0.2088
9x9F01	0.9369	0.9326	0.0008	0.2954
9x9F02	0.9424	0.9380	0.0008	0.2088
10x10A01	0.9377	0.9335	0.0008	0.3170
10x10A02	0.9426	0.9386	0.0007	0.2159
10x10A03	0.9396	0.9356	0.0007	0.3169
B10x10A01	0.9457	0.9414	0.0008	0.2212
10x10B01	0.9384	0.9341	0.0008	0.2881
10x10B02	0.9416	0.9373	0.0008	0.2333

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68				
Fuel Assembly Designation	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
10x10B03	0.9375	0.9334	0.0007	0.2856
B10x10B01	0.9436	0.9395	0.0007	0.2366
10x10C01	0.9433	0.9392	0.0007	0.2416
10x10D01	0.9376	0.9333	0.0008	0.3355
10x10E01	0.9185	0.9144	0.0007	0.2936

MPC-24 400PPM SOLUBLE BORON				
Fuel Assembly Designation	***Maximum k_{eff}***	***Calculated k_{eff}***	***Std. Dev. (1-sigma)***	***EALF (eV)***
14x14A03	*0.8884*	*0.8841*	*0.0008*	*0.2501*
B14x14B01	*0.8900*	*0.8855*	*0.0009*	*0.3173*
14x14C03	*0.8950*	*0.8907*	*0.0008*	*0.3410*
14x14D01	*0.8518*	*0.8475*	*0.0008*	*0.4395*
15x15A01	*0.9119*	*0.9076*	*0.0008*	*0.3363*
B15x15B01	*0.9284*	*0.9241*	*0.0008*	*0.3398*
B15x15C01	*0.9236*	*0.9193*	*0.0008*	*0.3074*
15x15D04	*0.9261*	*0.9218*	*0.0008*	*0.3841*
15x15E01	*0.9265*	*0.9221*	*0.0008*	*0.3656*
15x15F01	*0.9314*	*0.9271*	*0.0008*	*0.3791*
15x15G01	*0.8939*	*0.8897*	*0.0007*	*0.4392*
15x15H01	*0.9366*	*0.9320*	*0.0009*	*0.3175*
16x16A03	*0.8993*	*0.8948*	*0.0009*	*0.3164*
17x17A01	*0.9264*	*0.9221*	*0.0008*	*0.2801*

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-24 400PPM SOLUBLE BORON				
Fuel Assembly Designation	***Maximum k_{eff}***	***Calculated k_{eff}***	***Std. Dev. (1-sigma)***	***EALF (eV)***
17x17B06	*0.9284*	*0.9241*	*0.0008*	*0.3383*
17x17C02	*0.9294*	*0.9249*	*0.0009*	*0.3433*

MPC-32, 4.1% Enrichment, Bounding Cases				
Fuel Assembly Designation	***Maximum k_{eff}***	***Calculated k_{eff}***	***Std. Dev. (1-sigma)***	***EALF (eV)***
14x14A03	*0.9041*	*0.9001*	*0.0006*	*0.3185*
B14x14B01	*0.9257*	*0.9216*	*0.0007*	*0.4049*
14x14C01	*0.9423*	*0.9382*	*0.0007*	*0.4862*
14x14D01	*0.8970*	*0.8931*	*0.0006*	*0.5474*
15x15A01	*0.9206*	*0.9167*	*0.0006*	*0.5072*
B15x15B01	*0.9397*	*0.9358*	*0.0006*	*0.4566*
B15x15C01	*0.9266*	*0.9227*	*0.0006*	*0.4167*
15x15D04	*0.9384*	*0.9345*	*0.0006*	*0.5594*
15x15E01	*0.9365*	*0.9326*	*0.0006*	*0.5403*
15x15F01	*0.9411*	*0.9371*	*0.0006*	*0.4923*
15x15G01	*0.9147*	*0.9108*	*0.0006*	*0.5880*
15x15H01	*0.9276*	*0.9237*	*0.0006*	*0.4710*
16x16A03	*0.9375*	*0.9333*	*0.0007*	*0.4488*
17x17A01	*0.9111*	*0.9072*	*0.0006*	*0.4055*
17x17B06	*0.9309*	*0.9269*	*0.0006*	*0.4365*
17x17C02	*0.9355*	*0.9317*	*0.0006*	*0.4469*

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-32, 5.0% Enrichment, Bounding Cases				
Fuel Assembly Designation	***Maximum k_{eff}***	***Calculated k_{eff}***	***Std. Dev. (1-sigma)***	***EALF (eV)***
14x14A03	*0.9000*	*0.8959*	*0.0007*	*0.4651*
B14x14B01	*0.9214*	*0.9175*	*0.0006*	*0.6009*
14x14C01	*0.9480*	*0.9440*	*0.0006*	*0.6431*
14x14D01	*0.9050*	*0.9009*	*0.0007*	*0.7276*
15x15A01	*0.9230*	*0.9189*	*0.0007*	*0.7143*
B15x15B01	*0.9429*	*0.9390*	*0.0006*	*0.7234*
B15x15C01	*0.9307*	*0.9268*	*0.0006*	*0.6439*
15x15D04	*0.9466*	*0.9425*	*0.0007*	*0.7525*
15x15E01	*0.9434*	*0.9394*	*0.0007*	*0.7215*
15x15F01	*0.9483*	*0.9443*	*0.0007*	*0.7426*
15x15G01	*0.9251*	*0.9212*	*0.0006*	*0.9303*
15x15H01	*0.9333*	*0.9292*	*0.0007*	*0.7015*
16x16A03	*0.9429*	*0.9388*	*0.0007*	*0.5920*
17x17A01	*0.9161*	*0.9122*	*0.0006*	*0.6141*
17x17B06	*0.9371*	*0.9331*	*0.0006*	*0.6705*
17x17C02	*0.9437*	*0.9399*	*0.0006*	*0.6780*

Note: Maximum k_{eff} = Calculated k_{eff} + $K_c \times \sigma_c$ + Bias + σ_B
where:
K_c = 2.0
σ_c = Std. Dev. (1-sigma)
Bias = 0.0021
σ_B = 0.0006

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

See Subsection 6.4.3 for further explanation.

APPENDIX 6.D: SAMPLE INPUT FILES

(Total number of pages in this appendix : Error! Bookmark not defined.**)**

```
c
c Bounding Assembly in Class 15x15F
c
c MPC-24/24E cell configuration
c
c HI-STAR with active length 150 inch
c
c Cask Input Preprocessor
c cskinp 15f 15f mpc24n mpc24n histar star150 4.1 4rf5f45 pure
c ------     cpp\15f.bat
c   added 15f.co
c   added 15f.ce
c   added 15f.su
c   added 15f.sp
c ------     cpp\mpc24n.bat
c   added mpc24n.co
c   added mpc24n.ce
c   added mpc24n.su
c   added mpc24n.sp
c ------      cpp\histar.bat
c   added histar.co
c   added histar.ce
c   added histar.su
c   added histar.sp
c end of comments
c
c start of cells
c
c 15x15f
c
c number of cells: 6
c cell numbers:      1 to 7 and 201 to 299
c univers numbers:   1 to 3 and 201 to 299
c surface numbers:   1 to 9 and 201 to 299
c
c number of cells: 1
1     1 -10.522     -1  u=2       $ fuel
2     4 -1.0     1  -2  u=2       $ gap
3     3 -6.55    2  -3  u=2       $ Zr Clad
4     4 -1.0        3  u=2       $ water in fuel region
5     4 -1.0  -4:5     u=3       $ water in guide tubes
6     3 -6.55   4 -5    u=3       $ guide tubes
7     4 -1.0  -6   +7   -8   +9   u=1  lat=1
      fill= -8:8    -8:8   0:0
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 2 2 2 3 2 2 2 3 2 2 2 2 2 1
      1 2 2 2 3 2 2 2 2 2 2 2 3 2 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 3 2 2 3 2 2 2 3 2 2 3 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 2 2 2 2 2 3 2 2 2 2 2 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
```

```
      1 2 2 3 2 2 3 2 2 2 3 2 2 3 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 2 3 2 2 2 2 2 2 2 3 2 2 2 1
      1 2 2 2 2 2 3 2 2 2 3 2 2 2 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
c
c     MPC-24
c
c number of cells: 102
c cell numbers :      400   to  699
c universe numbers :    4   to    9
c surface numbers :   400   to  699
c
c     Right Side
c
408     0          -410   411  -412   413    u=4 fill=1 (1)
409     5  -7.84    410  -424   413  -426    u=4
410    4 -1.0     424  -428   448  -445    u=4
411    7  -2.7     428  -528  448  -445    u=4
412    6  -2.66   528  -532  448  -445    u=4
413    7  -2.7    532  -432   448  -445    u=4
414    4 -1.0     432  -436   448  -445    u=4
415    5  -7.84    436  -440   448  -445    u=4
416    4 -1.0     440        413         u=4
417    4 -1.0     424  -440   413  -447    u=4
418    4 -1.0     424  -440   446          u=4
419    5  -7.84    424  -440   447  -448    u=4
420    5  -7.84    424  -440   445  -446    u=4
c
c    Left Side
c
421    5  -7.84    425  -411   413          u=4
422    4 -1.0     429  -425   448  -445    u=4
423    7  -2.7    529  -429   448  -445    u=4
424    6  -2.66   533 -529   448  -445    u=4
425    7  -2.7     433 -533   448  -445    u=4
426    4 -1.0     437  -433   448  -445    u=4
427    5  -7.84    441  -437   448  -445    u=4
428    4 -1.0        -441   413         u=4
429    4 -1.0     441  -425   413  -447    u=4
430    4 -1.0     441  -425   446          u=4
431    5  -7.84    441  -425   447  -448    u=4
432    5  -7.84    441  -425   445  -446    u=4
c
c     Top
c
433    5  -7.84    411  -410   412  -426    u=4
434    4 -1.0     451  -452   426  -430    u=4
435    7  -2.7     451  -452   430 -530    u=4
436    6  -2.66    451  -452  530 -534    u=4
437    7  -2.7     451  -452  534  -434    u=4
438    4 -1.0     451  -452   434  -438    u=4
439    5  -7.84    451  -452   438  -442    u=4
```

```
440    4 -1.0     411  -424   442          u=4
441    4 -1.0     411  -450   426  -442    u=4
442    4 -1.0     453  -424   426  -442    u=4
443    5  -7.84    450  -451   426  -442    u=4
444    5  -7.84    452  -453   426  -442    u=4
c
c     Bottom
c
445    5  -7.84    427            -413    u=4
446    4 -1.0     451  -452   431  -427    u=4
447    7  -2.7     451  -452  531  -431    u=4
448    6  -2.66    451  -452  535 -531    u=4
449    7  -2.7     451  -452   435 -535    u=4
450    4 -1.0     451  -452   439  -435    u=4
451    5  -7.84    451  -452   443  -439    u=4
452    4 -1.0     411            -443    u=4
453    4 -1.0     411  -450   443  -427    u=4
454    4 -1.0     453        443  -427    u=4
455    5  -7.84    450  -451   443  -427    u=4
456    5  -7.84    452  -453   443  -427    u=4
457    5  -7.84    425  -411       -427    u=4
458    4 -1.0         -425       -427    u=4
c
c     TYPE B CELL  -  Short Boral on top and right
c
c     Right Side
c
459    0          -410   411  -412   413    u=5 fill=1 (1)
460    5  -7.84    410  -424   413  -426    u=5
470    4 -1.0     424  -428  548 -545    u=5
471    7  -2.7     428  -528 548 -545    u=5
472    6  -2.66   528  -532 548 -545    u=5
473    7  -2.7    532  -432  548 -545    u=5
474    4 -1.0     432  -436  548 -545    u=5
475    5  -7.84    436  -440  548 -545    u=5
476    4 -1.0     440        413         u=5
477    4 -1.0     424  -440   413 -547    u=5
478    4 -1.0     424  -440  546         u=5
479    5  -7.84    424  -440  547 -548    u=5
480    5  -7.84    424  -440  545 -546    u=5
c
c    Left Side
c
481    5  -7.84    425  -411   413         u=5
482    4 -1.0     429  -425   448  -445    u=5
483    7  -2.7     529  -429   448  -445    u=5
484    6  -2.66   533 -529   448  -445    u=5
485    7  -2.7     433 -533   448  -445    u=5
486    4 -1.0     437  -433   448  -445    u=5
487    5  -7.84    441  -437   448  -445    u=5
488    4 -1.0         -441   413         u=5
489    4 -1.0     441  -425   413  -447    u=5
490    4 -1.0     441  -425   446         u=5
491    5  -7.84    441  -425   447  -448    u=5
492    5  -7.84    441  -425   445  -446    u=5
```

```
c
c     Top
c
493    5  -7.84    411  -410   412  -426    u=5
494    4 -1.0    551 -552   426  -430    u=5
495    7  -2.7    551 -552   430 -530    u=5
496    6  -2.66   551 -552  530 -534    u=5
497    7  -2.7    551 -552  534  -434    u=5
498    4 -1.0    551 -552   434  -438    u=5
499    5  -7.84   551 -552   438  -442    u=5
500   4 -1.0     411  -424   442         u=5
501   4 -1.0     411 -550   426  -442    u=5
502   4 -1.0    553  -424   426  -442    u=5
503   5  -7.84   550 -551   426  -442    u=5
504   5  -7.84   552 -553   426  -442    u=5
c
c     Bottom
c
505   5  -7.84    427            -413    u=5
506   4 -1.0     451  -452   431  -427    u=5
507   7  -2.7     451  -452  531  -431    u=5
508   6  -2.66    451  -452  535 -531    u=5
509   7  -2.7     451  -452   435 -535    u=5
510   4 -1.0     451  -452   439  -435    u=5
511   5  -7.84    451  -452   443  -439    u=5
512   4 -1.0     411            -443    u=5
513   4 -1.0     411  -450   443  -427    u=5
514   4 -1.0     453        443  -427    u=5
515   5  -7.84    450  -451   443  -427    u=5
516   5  -7.84    452  -453   443  -427    u=5
517   5  -7.84    425  -411       -427    u=5
518   4 -1.0         -425       -427    u=5
c
c
c
c     TYPE D CELL  -  Short Boral on left and bottom, different cell ID
c
c number of cells: 51
c
c     Right Side
c
1570   0          -1410   1411  -1412   1413    u=17 fill=1 (1)
1571   5  -7.84    1410  -1424   1413  -1426    u=17
1572   4 -1.0     1424  -1428   1448  -1445    u=17
1573   7  -2.7     1428  -1528  1448  -1445    u=17
1574   6  -2.66   1528  -1532  1448  -1445    u=17
1575   7  -2.7    1532  -1432   1448  -1445    u=17
1576   4 -1.0     1432  -1436   1448  -1445    u=17
1577   5  -7.84    1436  -1440   1448  -1445    u=17
1578   4 -1.0     1440        1413         u=17
1579   4 -1.0     1424  -1440   1413  -1447    u=17
1580   4 -1.0     1424  -1440   1446         u=17
1581   5  -7.84    1424  -1440   1447  -1448    u=17
1582   5  -7.84    1424  -1440   1445  -1446    u=17
c
```

```
c    Left Side
c
1583   5  -7.84    1425  -1411   1413         u=17
1584   4 -1.0     1429  -1425  1548 -1545    u=17
1585   7  -2.7    1529  -1429  1548 -1545    u=17
1586   6  -2.66   1533 -1529  1548 -1545    u=17
1587   7  -2.7     1433 -1533  1548 -1545    u=17
1588   4 -1.0     1437  -1433  1548 -1545    u=17
1589   5  -7.84    1441  -1437  1548 -1545    u=17
1590   4 -1.0         -1441   1413         u=17
1591   4 -1.0     1441  -1425   1413 -1547    u=17
1592   4 -1.0     1441  -1425  1546         u=17
1593   5  -7.84    1441  -1425  1547 -1548    u=17
1594   5  -7.84    1441  -1425  1545 -1546    u=17
c
c     Top
c
1595   5  -7.84    1411  -1410   1412  -1426    u=17
1596   4 -1.0     1451  -1452   1426  -1430    u=17
1597   7  -2.7     1451  -1452   1430 -1530    u=17
1598   6  -2.66    1451  -1452  1530 -1534    u=17
1599   7  -2.7     1451  -1452  1534  -1434    u=17
1600   4 -1.0     1451  -1452   1434  -1438    u=17
1601   5  -7.84    1451  -1452   1438  -1442    u=17
1602   4 -1.0     1411  -1424   1442         u=17
1603   4 -1.0     1411  -1450   1426  -1442    u=17
1604   4 -1.0     1453  -1424   1426  -1442    u=17
1605   5  -7.84    1450  -1451   1426  -1442    u=17
1606   5  -7.84    1452  -1453   1426  -1442    u=17
c
c     Bottom
c
1607   5  -7.84    1427            -1413    u=17
1608   4 -1.0    1551 -1552   1431  -1427    u=17
1609   7  -2.7    1551 -1552  1531  -1431    u=17
1610   6  -2.66   1551 -1552  1535 -1531    u=17
1611   7  -2.7    1551 -1552   1435 -1535    u=17
1612   4 -1.0    1551 -1552   1439  -1435    u=17
1613   5  -7.84   1551 -1552   1443  -1439    u=17
1614   4 -1.0     1411            -1443    u=17
1615   4 -1.0     1411 -1550   1443  -1427    u=17
1616   4 -1.0    1553        1443  -1427    u=17
1617   5  -7.84   1550 -1551   1443  -1427    u=17
1618   5  -7.84   1552 -1553   1443  -1427    u=17
1619   5  -7.84    1425  -1411       -1427    u=17
1620   4 -1.0         -1425       -1427    u=17
c
c number of cells: 29
c
c empty cell no boral, no top
c
c
751     4 -1.0   -410   411  -412   413    u=14
752     5  -7.84    410  -424   413  -426    u=14
753    5  -7.84    425  -411   413         u=14
```

```
754    4 -1.0     411  -410   412  -426    u=14
755    5  -7.84    427            -413    u=14
756    5  -7.84    425  -411      -427    u=14
757    4 -1.0     411   426              u=14
758    4 -1.0     411  -427              u=14
759    4 -1.0    -425   413              u=14
760    4 -1.0     424   413   -426       u=14
761    4 -1.0    -425  -427              u=14
c
c
701   5   -7.84   701 -702 711 -713     u=9   $ steel post
702   5   -7.84   702 -703 711 -712     u=9   $ steel post
c
711   0           701 -705 711 -715 (702:713) (703:712)
                  fill=4 (13.8506 13.8506 0) u=9
712   0           704 (-706:-716) (705:715) -717 -710
                  fill=4 (17.9489 41.5518 0  0 1 0  -1 0 0   0 0 1) u=9
713   0           (705:715) -707 714 (-706:-716)  710
                  fill=4 (41.5518 17.9489 0  0 -1 0  1 0 0   0 0 1) u=9
714   0           701 -705 717 -719
                  fill=5 (13.8506 69.253 0) u=9
715   0           707 -709 711 -715
                  fill=5 (69.253 13.8506 0) u=9
716   0           706 -708 716 -718
                  fill=17 (45.6501 45.6501 0  -1 0 0  0 -1 0  0 0 1) u=9
717   0           705 -706 717 -719
                  fill=14 (41.5518 69.253 0) u=9
718   0           707 -709 715 -716
                  fill=14 (69.253 41.5518 0 0 1 0   1 0 0   0 0 1) u=9
719   0           701 -704 715 -717
                  fill=14 (-9.75233 41.5518 0  -1 0 0  0 1 0   0 0 1) u=9
720   0           705 -707 711 -714
                  fill=14 (41.5518 -9.75233 0   0 -1 0  1 0 0   0 0 1) u=9
721   4 -1.0     (706:719) (708:718) (709:716) u=9
c
c
c q-offset  0 inch
c
731   4 -1.0      720  721   fill=9 (0 0 0) u=19
732   4 -1.0     -720  721   fill=9 (0 0 0
                              -1 0 0  0  1 0  0 0 1) u=19
733   4 -1.0      720 -721   fill=9 (0 0 0
                               1 0 0  0 -1 0  0 0 1) u=19
734   4 -1.0     -720 -721   fill=9 (0 0 0
                              -1 0 0  0 -1 0  0 0 1) u=19
c
673   0           -41               39  -40  fill=19
c
c number of cells: 6
102   4  -1.0    -41       40  -44             $ 6.0" Water above Fuel
103   5  -7.84   -41       44  -45             $ 15.5" Steel above Fuel
104   4  -1.0    -41      -39   43             $ 7.3" Water below Fuel
105   5  -7.84   -41      -43   46             $ 8.5" Steel below Fuel
106   5  -7.84   46  -45 41 -42                $ 6.0" Radial Steel Shield
107   0            -46:45:42                   $ Outside world
```

```
c end of cells
c --- empty line

c --- empty line
c start of surfaces
1     cz         0.4752    $ fuel
2     cz         0.4851    $ clad ID
3     cz         0.5436    $ clad OD
4     cz         0.6350    $ guide ID
5     cz         0.6706    $ guide OD
6     px         0.7214    $ pin pitch
7     px        -0.7214
8     py         0.7214
9     py        -0.7214
c
c
c cell-id      8.98
c cell-pitch   10.906
c wall-thkns   5/16
c angle-thkns  5/16
c boral-gap    0.0035
c boral-gap-o  0.0035
c boral-thkns  0.075
c boral-clad   0.01
c sheathing    0.0235
c boral-wide   7.5
c boral-narrow 6.25
c
c gap size     1.09
c basket-od    67.335
c
410    px         11.40460 $x 8.98/2
411    px        -11.40460 $x {410} *-1
412    py         11.40460 $x {410}
413    py        -11.40460 $x {411}
424    px         12.19835 $x {410} + 5/16     $ angle
425    px        -12.19835 $x {411} - 5/16     $ box wall
426    py         12.19835 $x {412} + 5/16
427    py        -12.19835 $x {413} - 5/16
428    px         12.20724 $x {424} + 0.0035   $ wall to boral gap
429    px        -12.20724 $x {425} - 0.0035
430    py         12.20724 $x {426} + 0.0035
431    py        -12.20724 $x {427} - 0.0035
432    px         12.39774 $x {428} + 0.075    $ boral
433    px        -12.39774 $x {429} - 0.075
434    py         12.39774 $x {430} + 0.075
435    py        -12.39774 $x {431} - 0.075
436    px         12.40663 $x {432} + 0.0035   $ boral to sheathing gap
437    px        -12.40663 $x {433} - 0.0035
438    py         12.40663 $x {434} + 0.0035
439    py        -12.40663 $x {435} - 0.0035
440    px         12.46632 $x {436} + 0.0235   $ sheathing
441    px        -12.46632 $x {437} - 0.0235
442    py         12.46632 $x {438} + 0.0235
443    py        -12.46632 $x {439} - 0.0235
```

```
445    py          9.52500 $x 7.5/2
446    py          9.58469 $x {445} + 0.0235   $ sheathing
447    py         -9.58469 $x {446} *-1
448    py         -9.52500 $x {445} *-1
450    px         -9.58469 $x {447}
451    px         -9.52500 $x {448}
452    px          9.52500 $x {445}
453    px          9.58469 $x {446}
528   px        12.23264 $x {428} + 0.01  $ Aluminum on the outside of boral
529   px       -12.23264 $x {429} - 0.01
530   py        12.23264 $x {430} + 0.01
531   py       -12.23264 $x {431} - 0.01
532   px        12.37234 $x {432} - 0.01
533   px       -12.37234 $x {433} + 0.01
534   py        12.37234 $x {434} - 0.01
535   py       -12.37234 $x {435} + 0.01
545   py        7.93750 $x 6.25/2
546   py        7.99719 $x {545} + 0.0235   $ sheathing
547   py       -7.99719 $x {546} *-1
548   py       -7.93750 $x {545} *-1
550   px       -7.99719 $x {547}
551   px       -7.93750 $x {548}
552   px        7.93750 $x {545}
553   px        7.99719 $x {546}
c
c cell-id-2 8.98
c gap-o     1.09
c
701   px       -5.0
702   px       1.90627 $x (10.906 - 8.98)/2 - 5/16 + 0.1
703   px       3.45694 $x 2.722/2
704   px       4.09829 $x 10.906 - 8.98 - 5/16
705   px       27.70124 $x 10.906
706   px       31.79953 $x 2 * 10.906 - (8.98+8.98)/2 - 5/16
707   px       55.40248 $x 2 * 10.906
708   px       59.50077 $x {707} + {704}
709   px       83.10372 $x 3 * 10.906
710   p     1  -1  0  0.1 $ diagonal x=y, offset by 0.1 to avoid intersecting corners
711   py       -4.99999 $x {701}
712   py       1.90627 $x {702}
713   py       3.45694 $x {703}
714   py       4.09829 $x {704}
715   py       27.70124 $x {705}
716   py       31.79953 $x {706}
717   py       55.40248 $x {707}
718   py       59.50077 $x {708}
719   py       83.10372 $x {709}
720   px       0.0
721   py       0.0
1410    px         11.40460 $x 8.98/2
1411    px        -11.40460 $x {1410} *-1
1412    py         11.40460 $x {1410}
1413    py        -11.40460 $x {1411}
1424    px         12.19835 $x {1410} + 5/16     $ angle
1425    px        -12.19835 $x {1411} - 5/16     $ box wall
```

```
1426    py         12.19835 $x {1412} + 5/16
1427    py        -12.19835 $x {1413} - 5/16
1428    px         12.20724 $x {1424} + 0.0035   $ wall to boral gap
1429    px        -12.20724 $x {1425} - 0.0035
1430    py         12.20724 $x {1426} + 0.0035
1431    py        -12.20724 $x {1427} - 0.0035
1432    px         12.39774 $x {1428} + 0.075    $ boral
1433    px        -12.39774 $x {1429} - 0.075
1434    py         12.39774 $x {1430} + 0.075
1435    py        -12.39774 $x {1431} - 0.075
1436    px         12.40663 $x {1432} + 0.0035   $ boral to sheathing gap
1437    px        -12.40663 $x {1433} - 0.0035
1438    py         12.40663 $x {1434} + 0.0035
1439    py        -12.40663 $x {1435} - 0.0035
1440    px         12.46632 $x {1436} + 0.0235   $ sheathing
1441    px        -12.46632 $x {1437} - 0.0235
1442    py         12.46632 $x {1438} + 0.0235
1443    py        -12.46632 $x {1439} - 0.0235
1445    py          9.52500 $x 7.5/2
1446    py          9.58469 $x {1445} + 0.0235   $ sheathing
1447    py         -9.58469 $x {1446} *-1
1448    py         -9.52500 $x {1445} *-1
1450    px         -9.58469 $x {1447}
1451    px         -9.52500 $x {1448}
1452    px          9.52500 $x {1445}
1453    px          9.58469 $x {1446}
1528   px         12.23264 $x {1428} + 0.01  $ Aluminum on the outside of boral
1529   px        -12.23264 $x {1429} - 0.01
1530   py         12.23264 $x {1430} + 0.01
1531   py        -12.23264 $x {1431} - 0.01
1532   px         12.37234 $x {1432} - 0.01
1533   px        -12.37234 $x {1433} + 0.01
1534   py         12.37234 $x {1434} - 0.01
1535   py        -12.37234 $x {1435} + 0.01
1545   py         7.93750 $x 6.25/2
1546   py         7.99719 $x {1545} + 0.0235   $ sheathing
1547   py        -7.99719 $x {1546} *-1
1548   py        -7.93750 $x {1545} *-1
1550   px        -7.99719 $x {1547}
1551   px        -7.93750 $x {1548}
1552   px         7.93750 $x {1545}
1553   px         7.99719 $x {1546}
46    pz         -31.75       $ 8.5"  lower steel thickness
43    pz         -10.16       $ lower water thickness
39    pz           0.0        $ bottom of active fuel assembly
40    pz         381.0        $ top of active fuel assembly
44    pz         396.24       $ upper water thickness
45    pz         435.61       $ 15.5" upper steel thickness
41    cz          85.57       $ mpc steel ID
42    cz         108.43       $ mpc water
c end of surfaces
c --- empty line

c --- empty line
tr1 0 0 0
```

```
kcode    10000  .94  20  120
sdef  par=1  erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1  -2 1.2895
c
sp3  0 1
c
si4  s         13 14
            12 13 14 15
         11 12 13 14 15 16
         11 12 13 14 15 16
            12 13 14 15
               13 14
sp4  1 23r
c
ds5  s               26 26
                  25 25 25 25
               24 24 24 24 24 24
               23 23 23 23 23 23
                  22 22 22 22
                     21 21
c
si11  -79.25435 -57.61355
si12  -51.88077 -30.23997
si13  -24.50719 -2.86639
si14   2.86639   24.50719
si15   30.23997  51.88077
si16   57.61355  79.25435
c
si21  -79.25435 -57.61355
si22  -51.88077 -30.23997
si23  -24.50719 -2.86639
si24   2.86639   24.50719
si25   30.23997  51.88077
si26   57.61355  79.25435
c
sp11  0 1
sp12  0 1
sp13  0 1
sp14  0 1
sp15  0 1
sp16  0 1
sp21  0 1
sp22  0 1
sp23  0 1
sp24  0 1
sp25  0 1
sp26  0 1
c
m3           40000.56c   1.           $ Zr Clad
m4           1001.50c    0.6667       $ Water
             8016.50c    0.3333
m5           24000.50c   0.01761      $ Steel
             25055.50c   0.001761
             26000.55c   0.05977
```

```
              28000.50c   0.008239
m6            5010.50c   -0.054427     $ Boral Central Section @ 0.02 g/cmsq
              5011.50c   -0.241373
              13027.50c  -0.6222
              6000.50c   -0.0821
m7            13027.50c   1.0
mt4          lwtr.01t
prdmp    j   -120   j   2
fm4      1000   1   -6
f4:n   1
sd4      1000
e4    1.000E-11   1.000E-10   5.000E-10   7.500E-10   1.000E-09   1.200E-09
      1.500E-09   2.000E-09   2.500E-09   3.000E-09
      4.700E-09   5.000E-09   7.500E-09   1.000E-08   2.530E-08
      3.000E-08   4.000E-08   5.000E-08   6.000E-08   7.000E-08
      8.000E-08   9.000E-08   1.000E-07   1.250E-07   1.500E-07
      1.750E-07   2.000E-07   2.250E-07   2.500E-07   2.750E-07
      3.000E-07   3.250E-07   3.500E-07   3.750E-07   4.000E-07
      4.500E-07   5.000E-07   5.500E-07   6.000E-07   6.250E-07
      6.500E-07   7.000E-07   7.500E-07   8.000E-07   8.500E-07
      9.000E-07   9.250E-07   9.500E-07   9.750E-07   1.000E-06
      1.010E-06   1.020E-06   1.030E-06   1.040E-06   1.050E-06
      1.060E-06   1.070E-06   1.080E-06   1.090E-06   1.100E-06
      1.110E-06   1.120E-06   1.130E-06   1.140E-06   1.150E-06
      1.175E-06   1.200E-06   1.225E-06   1.250E-06   1.300E-06
      1.350E-06   1.400E-06   1.450E-06   1.500E-06   1.590E-06
      1.680E-06   1.770E-06   1.860E-06   1.940E-06   2.000E-06
      2.120E-06   2.210E-06   2.300E-06   2.380E-06   2.470E-06
      2.570E-06   2.670E-06   2.770E-06   2.870E-06   2.970E-06
      3.000E-06   3.050E-06   3.150E-06   3.500E-06   3.730E-06
      4.000E-06   4.750E-06   5.000E-06   5.400E-06   6.000E-06
      6.250E-06   6.500E-06   6.750E-06   7.000E-06   7.150E-06
      8.100E-06   9.100E-06   1.000E-05   1.150E-05   1.190E-05
      1.290E-05   1.375E-05   1.440E-05   1.510E-05   1.600E-05
      1.700E-05   1.850E-05   1.900E-05   2.000E-05   2.100E-05
      2.250E-05   2.500E-05   2.750E-05   3.000E-05   3.125E-05
      3.175E-05   3.325E-05   3.375E-05   3.460E-05   3.550E-05
      3.700E-05   3.800E-05   3.910E-05   3.960E-05   4.100E-05
      4.240E-05   4.400E-05   4.520E-05   4.700E-05   4.830E-05
      4.920E-05   5.060E-05   5.200E-05   5.340E-05   5.900E-05
      6.100E-05   6.500E-05   6.750E-05   7.200E-05   7.600E-05
      8.000E-05   8.200E-05   9.000E-05   1.000E-04   1.080E-04
      1.150E-04   1.190E-04   1.220E-04   1.860E-04   1.925E-04
      2.075E-04   2.100E-04   2.400E-04   2.850E-04   3.050E-04
      5.500E-04   6.700E-04   6.830E-04   9.500E-04   1.150E-03
      1.500E-03   1.550E-03   1.800E-03   2.200E-03   2.290E-03
      2.580E-03   3.000E-03   3.740E-03   3.900E-03   6.000E-03
      8.030E-03   9.500E-03   1.300E-02   1.700E-02   2.500E-02
      3.000E-02   4.500E-02   5.000E-02   5.200E-02   6.000E-02
      7.300E-02   7.500E-02   8.200E-02   8.500E-02   1.000E-01
      1.283E-01   1.500E-01   2.000E-01   2.700E-01   3.300E-01
      4.000E-01   4.200E-01   4.400E-01   4.700E-01   4.995E-01
      5.500E-01   5.730E-01   6.000E-01   6.700E-01   6.790E-01
      7.500E-01   8.200E-01   8.611E-01   8.750E-01   9.000E-01
      9.200E-01   1.010E+00   1.100E+00   1.200E+00   1.250E+00
```

```
      1.317E+00   1.356E+00   1.400E+00   1.500E+00   1.850E+00
      2.354E+00   2.479E+00   3.000E+00   4.304E+00   4.800E+00
      6.434E+00   8.187E+00   1.000E+01   1.284E+01   1.384E+01
      1.455E+01   1.568E+01   1.733E+01   2.000E+01
si3  h 0 381.00
imp:n 1 193r 0
c fuel enrichment 4.1 %
m1      92235.50c     -0.03614
        92238.50c     -0.84536
         8016.50c     -0.11850
c end of file
c
```

```
HI-STAR containing MPC68, 08x08 @ 4.2 wt% Enrich.
c     4.20 % uniform enrichment,  unreflected cask, 0.0279 g/cmsq B-10 in Boral
c
c
1     1 -10.522     -1  u=2       $ fuel
2     4 -1.0     1  -2  u=2       $ gap
3     3 -6.55    2  -3  u=2       $ Zr Clad
4     4 -1.0         3  u=2       $ water in fuel region
5     4 -1.0   -4:5     u=3       $ water in guide tubes
6     3 -6.55   4 -5    u=3       $ guide tubes
7     4 -1.0   -6   +7   -8   +9   u=1  lat=1
      fill= -5:4    -5:4   0:0
      1  1 1 1 1 1 1 1 1  1
      1  2 2 2 2 2 2 2 2  1
      1  2 2 2 2 2 2 2 2  1
      1  2 2 2 2 2 2 2 2  1
      1  2 2 2 3 2 2 2 2  1
      1  2 2 2 2 3 2 2 2  1
      1  2 2 2 2 2 2 2 2  1
      1  2 2 2 2 2 2 2 2  1
      1  2 2 2 2 2 2 2 2  1
      1  1 1 1 1 1 1 1 1  1
c
C  BOX TYPE R
c
8     0  -10   11  -12   13         u=4 fill=1 (0.8128  0.8128  0)
9     3  -6.55    60   -61    62   -63  #8      u=4     $ Zr flow channel
10    4  -1.      64   -65    66   -67  #8 #9   u=4     $ water
11    5  -7.84    20   -23    67   -14          u=4     $ 0.075" STEEL
12    4  -1.      20   -23    14   -15          u=4     $ WATER POCKET
13    7  -2.7     20   -23    15   -16          u=4     $ Al CLAD
14    6  -2.66    20   -23    16   -17          u=4     $ BORAL Absorber
15    7  -2.7     20   -23    17   -18          u=4     $ Al Clad
16    4  -1.      20   -23    18  -118          u=4     $ Water
17    5  -7.84    118:-129:65:-66               u=4     $ Steel
18    4  -1.      64   -21    67   -118         u=4     $ Water
19    4  -1.      24   -65    67   -118         u=4     $ water
20    5  -7.84    21   -20    67   -118         u=4     $ Steel
21    5  -7.84    23   -24    67   -118         u=4     $ Steel
22    4  -1.     129   -64    33   -118         u=4     $ Water
c
23    5  -7.84    25   -64    30   -31          u=4     $ Steel
24    4  -1.      26   -25    30   -31          u=4     $ Water
25    7  -2.7     27   -26    30   -31          u=4     $ Al clad
26    6  -2.66    28   -27    30   -31          u=4     $ Boral
27    7  -2.7     29   -28    30   -31          u=4     $ Al clad
28    4  -1.     129   -29    30   -31          u=4     $ water
29    5  -7.84   129   -64    32   -30          u=4     $ Steel ends
30    5  -7.84   129   -64    31   -33          u=4     $ Steel ends
31    4  -1.     129   -64    66   -32          u=4     $ Water
c
c    Type A box - Boral only on left side
c
32    0  -10   11  -12   13         u=6 fill=1 (0.8128  0.8128  0)
33    3  -6.55    60   -61    62   -63  #8      u=6     $ Zr flow channel
```

```
34    4  -1.      64   -65    66  -118  #8 #9   u=6     $ water
35    5  -7.84    118:-129:65:-66               u=6     $ Steel
36    4  -1.     129   -64    67   -118         u=6     $ Water
c
37    5  -7.84    25   -64    30   -31          u=6     $ Steel
38    4  -1.      26   -25    30   -31          u=6     $ Water
39    7  -2.7     27   -26    30   -31          u=6     $ Al clad
40    6  -2.66    28   -27    30   -31          u=6     $ Boral
41    7  -2.7     29   -28    30   -31          u=6     $ Al clad
42    4  -1.     129   -29    30   -31          u=6     $ water
43    4  -1.     129   -64    33   -67          u=6     $ Water
44    5  -7.84   129   -64    32   -30          u=6     $ Steel ends
45    5  -7.84   129   -64    31   -33          u=6     $ Steel ends
46    4  -1.     129   -64    66   -32          u=6     $ Water
c
c    Type B box -  Boral on Top only
c
47    0  -10   11  -12   13         u=7 fill=1 (0.8128  0.8128  0)
48    3  -6.55    60   -61    62   -63  #8      u=7     $ Zr flow channel
49    4  -1.      64   -65    66   -67  #8 #9   u=7     $ water
50    5  -7.84    20   -23    67   -14          u=7     $ 0.075" STEEL
51    4  -1.      20   -23    14   -15          u=7     $ WATER POCKET
52    7  -2.7     20   -23    15   -16          u=7     $ Al CLAD
53    6  -2.66    20   -23    16   -17          u=7     $ BORAL Absorber
54    7  -2.7     20   -23    17   -18          u=7     $ water
55    4  -1.      20   -23    18  -118          u=7     $ Water
56    5  -7.84    118:-129:65:-66               u=7     $ Steel
57    4  -1.      64   -21    67   -118         u=7     $ Water
58    4  -1.      24   -65    67   -118         u=7     $ water
59    5  -7.84    21   -20    67   -118         u=7     $ Steel
60    5  -7.84    23   -24    67   -118         u=7     $ Steel
61    4  -1.     129   -64    66   -118         u=7     $ Water
c
c    Type E box  - No Boral Panels
c
62    0  -10   11  -12   13         u=8 fill=1 (0.8128  0.8128  0)
63    3  -6.55    60   -61    62   -63  #8      u=8     $ Zr flow channel
64    4  -1.     129   -65    66  -118  #8 #9   u=8     $ water
65    5  -7.84    118:-129:65:-66               u=8     $ Steel
c
c    Type F box  - No Boral Panels or fuel
c
66    4  -1.     129   -65    66  -118          u=9     $ water
67    5  -7.84    118:-129:65:-66               u=9     $ Steel
c
68    4  -1.0    -34  35  -36  37  u=5  lat=1  fill=-7:6  -7:6  0:0
               5 5 5 5 5 5 5 5 5 5 5 5 5 5
               5 9 9 9 9 9 9 9 9 9 9 9 9 5
               5 9 9 9 9 9 7 4 9 9 9 9 9 5
               5 9 9 9 7 4 4 4 4 4 9 9 9 5
               5 9 9 7 4 4 4 4 4 4 4 9 9 5
               5 9 9 7 4 4 4 4 4 4 4 9 9 5
               5 9 7 4 4 4 4 4 4 4 4 4 9 5
               5 9 8 4 4 4 4 4 4 4 4 6 9 5
               5 9 9 7 4 4 4 4 4 4 4 9 9 5
```

```
               5 9 9 8 4 4 4 4 4 4 6 9 9 5
               5 9 9 9 8 4 4 4 6 6 9 9 9 5
               5 9 9 9 9 9 8 6 9 9 9 9 9 5
               5 9 9 9 9 9 9 9 9 9 9 9 9 5
               5 5 5 5 5 5 5 5 5 5 5 5 5 5
69    0          -41              50  -49  fill=5 (8.1661  8.1661  0)
70    4  -1.0    -41       43  -50           $ Water below Fuel
71    4  -1.0    -41       49  -44           $ Water above Fuel
72    5  -7.84   -42       68  -43           $ Steel below Fuel
73    5  -7.84   -42       44  -69           $ Steel above Fuel
74    5  -7.84    41  -42  43  -44           $ Radial Steel
75    0            42 :-68:  69              $ outside world

1     cz          0.5207      $ Fuel OD
2     cz          0.5321      $ Clad ID
3     cz          0.6134      $ Clad OD
4     cz          0.6744      $ Thimble ID
5     cz          0.7506      $ Thimble OD
6     px          0.8128      $ Pin Pitch
7     px         -0.8128
8     py          0.8128
9     py         -0.8128
10    px          6.7031      $ Channel ID
11    px         -6.7031
12    py          6.7031
13    py         -6.7031
14    py          7.8016
15    py          7.8155
16    py          7.8410
17    py          8.0467
18    py          8.0721
118   py          8.0861
20    px         -6.0325
21    px         -6.2230
23    px          6.0325
24    px          6.2230
25    px         -7.8016
26    px         -7.8155
27    px         -7.8410
28    px         -8.0467
29    px         -8.0721
129   px         -8.0861
30    py         -6.0325
31    py          6.0325
32    py         -6.2230
33    py          6.2230
34    px          7.6111
35    px         -8.7211
36    py          8.7211
37    py         -7.6111
41    cz          85.57
42    cz          108.43
43    pz         -18.54
44    pz          402.5
49    pz          381.         $ Top of Active Fuel
```

```
50    pz           0          $ Start of Active Fuel
60    px         -6.9571      $ Channel OD
61    px          6.9571
62    py         -6.9571
63    py          6.9571
64    px         -7.6111      $ Cell Box ID
65    px          7.6111
66    py         -7.6111
67    py          7.6111
68    pz         -40.13
69    pz          441.9

imp:n     1 73r 0
kcode   10000  0.94   20  120
c
sdef  par=1  erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1 -2 1.2895
c
c
si3  h 0 365.76
sp3  0 1
c
c
si4  s                15 16
               13 14 15 16 17 18
            12 13 14 15 16 17 18 19
            12 13 14 15 16 17 18 19
         11 12 13 14 15 16 17 18 19 20
         11 12 13 14 15 16 17 18 19 20
            12 13 14 15 16 17 18 19
            12 13 14 15 16 17 18 19
               13 14 15 16 17 18
                     15 16
sp4  1 67r
c
ds5  s                30 30
               29 29 29 29 29 29
            28 28 28 28 28 28 28 28
            27 27 27 27 27 27 27 27
         26 26 26 26 26 26 26 26 26 26
         25 25 25 25 25 25 25 25 25 25
            24 24 24 24 24 24 24 24
            23 23 23 23 23 23 23 23
               22 22 22 22 22 22
                     21 21
c
si11  -80.6831 -67.6783
si12  -64.1985 -51.1937
si13  -47.7139 -34.7091
si14  -31.2293 -18.2245
si15  -14.7447 -1.7399
si16   1.7399   14.7447
si17   18.2245  31.2293
si18   34.7091  47.7139
```

```
si19   51.1937  64.1985
si20   67.6783  80.6831
c
si21  -80.6831 -67.6783
si22  -64.1985 -51.1937
si23  -47.7139 -34.7091
si24  -31.2293 -18.2245
si25  -14.7447 -1.7399
si26   1.7399   14.7447
si27   18.2245  31.2293
si28   34.7091  47.7139
si29   51.1937  64.1985
si30   67.6783  80.6831
sp11  0 1
sp12  0 1
sp13  0 1
sp14  0 1
sp15  0 1
sp16  0 1
sp17  0 1
sp18  0 1
sp19  0 1
sp20  0 1
sp21  0 1
sp22  0 1
sp23  0 1
sp24  0 1
sp25  0 1
sp26  0 1
sp27  0 1
sp28  0 1
sp29  0 1
sp30  0 1
c
m1           92235.50c   9.98343E-04  $ 4.20% E Fuel
             92238.50c   0.022484
             8016.50c    0.046965
m2           8016.50c    1.           $ Void
m3           40000.56c   1.           $ Zr Clad
m4           1001.50c    0.6667       $ Water
             8016.50c    0.3333
m5           24000.50c   0.01761      $ Steel
             25055.50c   0.001761
             26000.55c   0.05977
             28000.50c   0.008239
m6           5010.50c    8.0707E-03   $ Boral
             5011.50c    3.2553E-02
             6000.50c    1.0146E-02
             13027.50c   3.8054E-02
m7           13027.50c    1.           $ Al Clad
mt4         lwtr.01t
prdmp    j   -30   1   2
fm4      1000   1   -6
f4:n   1
sd4      1000
```

```
e4    1.000E-11   1.000E-10   5.000E-10   7.500E-10   1.000E-09   1.200E-09
      1.500E-09   2.000E-09   2.500E-09   3.000E-09
      4.700E-09   5.000E-09   7.500E-09   1.000E-08   2.530E-08
      3.000E-08   4.000E-08   5.000E-08   6.000E-08   7.000E-08
      8.000E-08   9.000E-08   1.000E-07   1.250E-07   1.500E-07
      1.750E-07   2.000E-07   2.250E-07   2.500E-07   2.750E-07
      3.000E-07   3.250E-07   3.500E-07   3.750E-07   4.000E-07
      4.500E-07   5.000E-07   5.500E-07   6.000E-07   6.250E-07
      6.500E-07   7.000E-07   7.500E-07   8.000E-07   8.500E-07
      9.000E-07   9.250E-07   9.500E-07   9.750E-07   1.000E-06
      1.010E-06   1.020E-06   1.030E-06   1.040E-06   1.050E-06
      1.060E-06   1.070E-06   1.080E-06   1.090E-06   1.100E-06
      1.110E-06   1.120E-06   1.130E-06   1.140E-06   1.150E-06
      1.175E-06   1.200E-06   1.225E-06   1.250E-06   1.300E-06
      1.350E-06   1.400E-06   1.450E-06   1.500E-06   1.590E-06
      1.680E-06   1.770E-06   1.860E-06   1.940E-06   2.000E-06
      2.120E-06   2.210E-06   2.300E-06   2.380E-06   2.470E-06
      2.570E-06   2.670E-06   2.770E-06   2.870E-06   2.970E-06
      3.000E-06   3.050E-06   3.150E-06   3.500E-06   3.730E-06
      4.000E-06   4.750E-06   5.000E-06   5.400E-06   6.000E-06
      6.250E-06   6.500E-06   6.750E-06   7.000E-06   7.150E-06
      8.100E-06   9.100E-06   1.000E-05   1.150E-05   1.190E-05
      1.290E-05   1.375E-05   1.440E-05   1.510E-05   1.600E-05
      1.700E-05   1.850E-05   1.900E-05   2.000E-05   2.100E-05
      2.250E-05   2.500E-05   2.750E-05   3.000E-05   3.125E-05
      3.175E-05   3.325E-05   3.375E-05   3.460E-05   3.550E-05
      3.700E-05   3.800E-05   3.910E-05   3.960E-05   4.100E-05
      4.240E-05   4.400E-05   4.520E-05   4.700E-05   4.830E-05
      4.920E-05   5.060E-05   5.200E-05   5.340E-05   5.900E-05
      6.100E-05   6.500E-05   6.750E-05   7.200E-05   7.600E-05
      8.000E-05   8.200E-05   9.000E-05   1.000E-04   1.080E-04
      1.150E-04   1.190E-04   1.220E-04   1.860E-04   1.925E-04
      2.075E-04   2.100E-04   2.400E-04   2.850E-04   3.050E-04
      5.500E-04   6.700E-04   6.830E-04   9.500E-04   1.150E-03
      1.500E-03   1.550E-03   1.800E-03   2.200E-03   2.290E-03
      2.580E-03   3.000E-03   3.740E-03   3.900E-03   6.000E-03
      8.030E-03   9.500E-03   1.300E-02   1.700E-02   2.500E-02
      3.000E-02   4.500E-02   5.000E-02   5.200E-02   6.000E-02
      7.300E-02   7.500E-02   8.200E-02   8.500E-02   1.000E-01
      1.283E-01   1.500E-01   2.000E-01   2.700E-01   3.300E-01
      4.000E-01   4.200E-01   4.400E-01   4.700E-01   4.995E-01
      5.500E-01   5.730E-01   6.000E-01   6.700E-01   6.790E-01
      7.500E-01   8.200E-01   8.611E-01   8.750E-01   9.000E-01
      9.200E-01   1.010E+00   1.100E+00   1.200E+00   1.250E+00
      1.317E+00   1.356E+00   1.400E+00   1.500E+00   1.850E+00
      2.354E+00   2.479E+00   3.000E+00   4.304E+00   4.800E+00
      6.434E+00   8.187E+00   1.000E+01   1.284E+01   1.384E+01
      1.455E+01   1.568E+01   1.733E+01   2.000E+01
```

```
HI-STAR containing MPC68F, 06x06 @ 3.0 wt% Enrich.
c     3.00 % uniform enrichment, unreflected cask, 0.0067 g/cmsq B-10 in Boral
c     Dresden-1 6x6
c
c
1     1 -10.522     -1  u=2       $ fuel
2     4 -1.0     1  -2  u=2       $ gap
3     3 -6.55    2  -3  u=2       $ Zr Clad
4     4 -1.0         3  u=2       $ water in fuel region
5     4 -1.0   -4:5     u=3       $ water in guide tubes
6     3 -6.55   4 -5    u=3       $ guide tubes
7     4 -1.0   -6   +7   -8   +9   u=1  lat=1
      fill= -4:3    -4:3   0:0
      1  1 1 1 1 1 1  1
      1  2 2 2 2 2 2  1
      1  2 2 2 2 2 2  1
      1  2 2 2 2 2 2  1
      1  2 2 2 2 2 2  1
      1  2 2 2 2 2 2  1
      1  2 2 2 2 2 2  1
      1  1 1 1 1 1 1  1
c
C  BOX TYPE R
c
8     0  -10   11  -12   13         u=4 fill=1 (0.8814  0.8814  0)
9     3  -6.55    60   -61    62   -63  #8      u=4     $ Zr flow channel
10    4  -1.      64   -65    66   -67  #8 #9   u=4     $ water
11    5  -7.84    20   -23    67   -14          u=4     $ 0.075" STEEL
12    4  -1.      20   -23    14   -15          u=4     $ WATER POCKET
13    7  -2.7     20   -23    15   -16          u=4     $ Al CLAD
14    6  -2.66    20   -23    16   -17          u=4     $ BORAL Absorber
15    7  -2.7     20   -23    17   -18          u=4     $ Al Clad
16    4  -1.      20   -23    18  -118          u=4     $ Water
17    5  -7.84    118:-129:65:-66               u=4     $ Steel
18    4  -1.      64   -21    67   -118         u=4     $ Water
19    4  -1.      24   -65    67   -118         u=4     $ water
20    5  -7.84    21   -20    67   -118         u=4     $ Steel
21    5  -7.84    23   -24    67   -118         u=4     $ Steel
22    4  -1.     129   -64    33   -118         u=4     $ Water
c
23    5  -7.84    25   -64    30   -31          u=4     $ Steel
24    4  -1.      26   -25    30   -31          u=4     $ Water
25    7  -2.7     27   -26    30   -31          u=4     $ Al clad
26    6  -2.66    28   -27    30   -31          u=4     $ Boral
27    7  -2.7     29   -28    30   -31          u=4     $ Al clad
28    4  -1.     129   -29    30   -31          u=4     $ water
29    5  -7.84   129   -64    32   -30          u=4     $ Steel ends
30    5  -7.84   129   -64    31   -33          u=4     $ Steel ends
31    4  -1.     129   -64    66   -32          u=4     $ Water
c
c    Type A box - Boral only on left side
c
32    0  -10   11  -12   13         u=6 fill=1 (0.8814  0.8814  0)
33    3  -6.55    60   -61    62   -63  #8      u=6     $ Zr flow channel
34    4  -1.      64   -65    66  -118  #8 #9   u=6     $ water
```

```
35    5  -7.84    118:-129:65:-66               u=6     $ Steel
36    4  -1.     129   -64    67   -118         u=6     $ Water
c
37    5  -7.84    25   -64    30   -31          u=6     $ Steel
38    4  -1.      26   -25    30   -31          u=6     $ Water
39    7  -2.7     27   -26    30   -31          u=6     $ Al clad
40    6  -2.66    28   -27    30   -31          u=6     $ Boral
41    7  -2.7     29   -28    30   -31          u=6     $ Al clad
42    4  -1.     129   -29    30   -31          u=6     $ water
43    4  -1.     129   -64    33   -67          u=6     $ Water
44    5  -7.84   129   -64    32   -30          u=6     $ Steel ends
45    5  -7.84   129   -64    31   -33          u=6     $ Steel ends
46    4  -1.     129   -64    66   -32          u=6     $ Water
c
c    Type B box -  Boral on Top only
c
47    0  -10   11  -12   13         u=7 fill=1 (0.8814  0.8814  0)
48    3  -6.55    60   -61    62   -63  #8      u=7     $ Zr flow channel
49    4  -1.      64   -65    66   -67  #8 #9   u=7     $ water
50    5  -7.84    20   -23    67   -14          u=7     $ 0.075" STEEL
51    4  -1.      20   -23    14   -15          u=7     $ WATER POCKET
52    7  -2.7     20   -23    15   -16          u=7     $ Al CLAD
53    6  -2.66    20   -23    16   -17          u=7     $ BORAL Absorber
54    7  -2.7     20   -23    17   -18          u=7     $ water
55    4  -1.      20   -23    18  -118          u=7     $ Water
56    5  -7.84    118:-129:65:-66               u=7     $ Steel
57    4  -1.      64   -21    67   -118         u=7     $ Water
58    4  -1.      24   -65    67   -118         u=7     $ water
59    5  -7.84    21   -20    67   -118         u=7     $ Steel
60    5  -7.84    23   -24    67   -118         u=7     $ Steel
61    4  -1.     129   -64    66   -118         u=7     $ Water
c
c    Type E box  - No Boral Panels
c
62    0  -10   11  -12   13         u=8 fill=1 (0.8814  0.8814  0)
63    3  -6.55    60   -61    62   -63  #8      u=8     $ Zr flow channel
64    4  -1.     129   -65    66  -118  #8 #9   u=8     $ water
65    5  -7.84    118:-129:65:-66               u=8     $ Steel
c
c    Type F box  - No Boral Panels or fuel
c
66    4  -1.     129   -65    66  -118          u=9     $ water
67    5  -7.84    118:-129:65:-66               u=9     $ Steel
c
68    4  -1.0    -34  35  -36  37  u=5  lat=1  fill=-7:6  -7:6  0:0
               5 5 5 5 5 5 5 5 5 5 5 5 5 5
               5 9 9 9 9 9 9 9 9 9 9 9 9 5
               5 9 9 9 9 9 7 4 9 9 9 9 9 5
               5 9 9 9 7 4 4 4 4 4 9 9 9 5
               5 9 9 7 4 4 4 4 4 4 4 9 9 5
               5 9 9 7 4 4 4 4 4 4 4 9 9 5
               5 9 7 4 4 4 4 4 4 4 4 4 9 5
               5 9 8 4 4 4 4 4 4 4 4 6 9 5
               5 9 9 7 4 4 4 4 4 4 4 9 9 5
               5 9 9 8 4 4 4 4 4 4 6 9 9 5
```

```
               5 9 9 9 8 4 4 4 6 6 9 9 9 5
               5 9 9 9 9 9 8 6 9 9 9 9 9 5
               5 9 9 9 9 9 9 9 9 9 9 9 9 5
               5 5 5 5 5 5 5 5 5 5 5 5 5 5
69    0           -41             50  -49  fill=5 (8.1661  8.1661  0)
70    4  -1.0    -41       43  -50           $ Water below Fuel
71    4  -1.0    -41       49  -44           $ Water above Fuel
72    5  -7.84   -42       68  -43           $ Steel below Fuel
73    5  -7.84   -42       44  -69           $ Steel above Fuel
74    5  -7.84    41  -42  43  -44           $ Radial Steel
75    0            42 :-68:  69              $ outside world

1     cz         0.6274       $ Fuel OD
2     cz         0.6280       $ Clad ID
3     cz         0.7169       $ Clad OD
4     cz         0.6280       $ Thimble ID
5     cz         0.7169       $ Thimble OD
6     px         0.8814       $ Pin Pitch
7     px        -0.8814
8     py         0.8814
9     py        -0.8814
10    px         5.4483       $ Channel ID
11    px        -5.4483
12    py         5.4483
13    py        -5.4483
14    py         7.8016
15    py         7.8155
16    py         7.8410
17    py         8.0467
18    py         8.0721
118   py         8.0861
20    px        -6.0325
21    px        -6.2230
23    px         6.0325
24    px         6.2230
25    px        -7.8016
26    px        -7.8155
27    px        -7.8410
28    px        -8.0467
29    px        -8.0721
129   px        -8.0861
30    py        -6.0325
31    py         6.0325
32    py        -6.2230
33    py         6.2230
34    px         7.6111
35    px        -8.7211
36    py         8.7211
37    py        -7.6111
41    cz         85.57
42    cz         108.43
43    pz         11.46
44    pz         331.0
49    pz         309.4         $ Top of Active F
50    pz          30.          $ Start of Active
```

```
60    px         -5.6007       $ Channel OD
61    px          5.6007
62    py         -5.6007
63    py          5.6007
64    px         -7.6111       $ Cell Box ID
65    px          7.6111
66    py         -7.6111
67    py          7.6111
68    pz         -10.13
69    pz          370.36

imp:n     1 73r 0
kcode    10000  0.94   20  120
c
sdef  par=1  erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1 -2 1.2895
c
si3  h 30. 309.
sp3  0 1
c
c
si4  s                15 16
                13 14 15 16 17 18
             12 13 14 15 16 17 18 19
             12 13 14 15 16 17 18 19
          11 12 13 14 15 16 17 18 19 20
          11 12 13 14 15 16 17 18 19 20
             12 13 14 15 16 17 18 19
             12 13 14 15 16 17 18 19
                13 14 15 16 17 18
                      15 16
sp4  1 67r
c
ds5  s                30 30
                29 29 29 29 29 29
             28 28 28 28 28 28 28 28
             27 27 27 27 27 27 27 27
          26 26 26 26 26 26 26 26 26 26
          25 25 25 25 25 25 25 25 25 25
             24 24 24 24 24 24 24 24
             23 23 23 23 23 23 23 23
                22 22 22 22 22 22
                      21 21
c
si11  -80.6831 -67.6783
si12  -64.1985 -51.1937
si13  -47.7139 -34.7091
si14  -31.2293 -18.2245
si15  -14.7447 -1.7399
si16   1.7399   14.7447
si17   18.2245  31.2293
si18   34.7091  47.7139
si19   51.1937  64.1985
si20   67.6783  80.6831
```

```
c
si21  -80.6831 -67.6783
si22  -64.1985 -51.1937
si23  -47.7139 -34.7091
si24  -31.2293 -18.2245
si25  -14.7447 -1.7399
si26   1.7399   14.7447
si27   18.2245  31.2293
si28   34.7091  47.7139
si29   51.1937  64.1985
si30   67.6783  80.6831
sp11  0 1
sp12  0 1
sp13  0 1
sp14  0 1
sp15  0 1
sp16  0 1
sp17  0 1
sp18  0 1
sp19  0 1
sp20  0 1
sp21  0 1
sp22  0 1
sp23  0 1
sp24  0 1
sp25  0 1
sp26  0 1
sp27  0 1
sp28  0 1
sp29  0 1
sp30  0 1
c
m1           92235.50c   -0.02644      $ 3.00% E Fuel
             92238.50c   -0.85504
             8016.50c    -0.11852
m3           40000.56c   1.           $ Zr Clad
m4           1001.50c    0.6667       $ Water
             8016.50c    0.3333
m5           24000.50c   0.01761      $ Steel
             25055.50c   0.001761
             26000.55c   0.05977
             28000.50c   0.008239
m6           5010.50c  1.9592E-03      $ Boral 0.0067 gm/cm2
             5011.50c  8.1175E-03
             6000.50c  2.5176E-03
             13027.50c 5.4933E-02
m7           13027.50c    1.           $ Al Clad
mt4         lwtr.01t
prdmp    j   -120   j   2
fm4      1000   1   -6
f4:n   1
sd4      1000
e4    1.000E-11   1.000E-10   5.000E-10   7.500E-10   1.000E-09   1.200E-09
      1.500E-09   2.000E-09   2.500E-09   3.000E-09
      4.700E-09   5.000E-09   7.500E-09   1.000E-08   2.530E-08
```

3.000E-08	4.000E-08	5.000E-08	6.000E-08	7.000E-08
8.000E-08	9.000E-08	1.000E-07	1.250E-07	1.500E-07
1.750E-07	2.000E-07	2.250E-07	2.500E-07	2.750E-07
3.000E-07	3.250E-07	3.500E-07	3.750E-07	4.000E-07
4.500E-07	5.000E-07	5.500E-07	6.000E-07	6.250E-07
6.500E-07	7.000E-07	7.500E-07	8.000E-07	8.500E-07
9.000E-07	9.250E-07	9.500E-07	9.750E-07	1.000E-06
1.010E-06	1.020E-06	1.030E-06	1.040E-06	1.050E-06
1.060E-06	1.070E-06	1.080E-06	1.090E-06	1.100E-06
1.110E-06	1.120E-06	1.130E-06	1.140E-06	1.150E-06
1.175E-06	1.200E-06	1.225E-06	1.250E-06	1.300E-06
1.350E-06	1.400E-06	1.450E-06	1.500E-06	1.590E-06
1.680E-06	1.770E-06	1.860E-06	1.940E-06	2.000E-06
2.120E-06	2.210E-06	2.300E-06	2.380E-06	2.470E-06
2.570E-06	2.670E-06	2.770E-06	2.870E-06	2.970E-06
3.000E-06	3.050E-06	3.150E-06	3.500E-06	3.730E-06
4.000E-06	4.750E-06	5.000E-06	5.400E-06	6.000E-06
6.250E-06	6.500E-06	6.750E-06	7.000E-06	7.150E-06
8.100E-06	9.100E-06	1.000E-05	1.150E-05	1.190E-05
1.290E-05	1.375E-05	1.440E-05	1.510E-05	1.600E-05
1.700E-05	1.850E-05	1.900E-05	2.000E-05	2.100E-05
2.250E-05	2.500E-05	2.750E-05	3.000E-05	3.125E-05
3.175E-05	3.325E-05	3.375E-05	3.460E-05	3.550E-05
3.700E-05	3.800E-05	3.910E-05	3.960E-05	4.100E-05
4.240E-05	4.400E-05	4.520E-05	4.700E-05	4.830E-05
4.920E-05	5.060E-05	5.200E-05	5.340E-05	5.900E-05
6.100E-05	6.500E-05	6.750E-05	7.200E-05	7.600E-05
8.000E-05	8.200E-05	9.000E-05	1.000E-04	1.080E-04
1.150E-04	1.190E-04	1.220E-04	1.860E-04	1.925E-04
2.075E-04	2.100E-04	2.400E-04	2.850E-04	3.050E-04
5.500E-04	6.700E-04	6.830E-04	9.500E-04	1.150E-03
1.500E-03	1.550E-03	1.800E-03	2.200E-03	2.290E-03
2.580E-03	3.000E-03	3.740E-03	3.900E-03	6.000E-03
8.030E-03	9.500E-03	1.300E-02	1.700E-02	2.500E-02
3.000E-02	4.500E-02	5.000E-02	5.200E-02	6.000E-02
7.300E-02	7.500E-02	8.200E-02	8.500E-02	1.000E-01
1.283E-01	1.500E-01	2.000E-01	2.700E-01	3.300E-01
4.000E-01	4.200E-01	4.400E-01	4.700E-01	4.995E-01
5.500E-01	5.730E-01	6.000E-01	6.700E-01	6.790E-01
7.500E-01	8.200E-01	8.611E-01	8.750E-01	9.000E-01
9.200E-01	1.010E+00	1.100E+00	1.200E+00	1.250E+00
1.317E+00	1.356E+00	1.400E+00	1.500E+00	1.850E+00
2.354E+00	2.479E+00	3.000E+00	4.304E+00	4.800E+00
6.434E+00	8.187E+00	1.000E+01	1.284E+01	1.384E+01
1.455E+01	1.568E+01	1.733E+01	2.000E+01	

```
HI-STAR containing MPC68F, 06x06 in DFC with 08 missing rods
c     3.00 % uniform enrichment, unreflected cask, 0.0067 g/cmsq B-10 in Boral
c
c
1     1 -10.522     -1  u=2       $ fuel
2     4 -1.0     1  -2  u=2       $ gap
3     3 -6.55    2  -3  u=2       $ Zr Clad
4     4 -1.0         3  u=2       $ water in fuel region
5     4 -1.0   -4:5     u=3       $ water in guide tubes
6     3 -6.55   4 -5    u=3       $ guide tubes
7     4 -1.0   -6   +7   -8   +9   u=1  lat=1
      fill= -4:3    -4:3   0:0
      1  1 1 1 1 1 1  1
      1  2 2 2 2 2 2  1
      1  2 1 2 1 2 2  1
      1  2 2 1 2 1 2  1
      1  2 1 2 1 2 2  1
      1  2 2 1 2 1 2  1
      1  2 2 2 2 2 2  1
      1  1 1 1 1 1 1  1
c
C  BOX TYPE R
c
8     0  -10   11  -12   13         u=4 fill=1 (0.8814  0.8814  0)
9     3  -6.55    60   -61    62   -63  #8      u=4     $ Zr flow channel
100   5  -7.84   74 -75 76 -77 (-70:71:-72:73)  u=4     $ DFC
10    4  -1.      64   -65    66   -67  #8 #9 #100  u=4 $ water
11    5  -7.84    20   -23    67   -14          u=4     $ 0.075" STEEL
12    4  -1.      20   -23    14   -15          u=4     $ WATER POCKET
13    7  -2.7     20   -23    15   -16          u=4     $ Al CLAD
14    6  -2.66    20   -23    16   -17          u=4     $ BORAL Absorber
15    7  -2.7     20   -23    17   -18          u=4     $ Al Clad
16    4  -1.      20   -23    18  -118          u=4     $ Water
17    5  -7.84    118:-129:65:-66               u=4     $ Steel
18    4  -1.      64   -21    67   -118         u=4     $ Water
19    4  -1.      24   -65    67   -118         u=4     $ water
20    5  -7.84    21   -20    67   -118         u=4     $ Steel
21    5  -7.84    23   -24    67   -118         u=4     $ Steel
22    4  -1.     129   -64    33   -118         u=4     $ Water
c
23    5  -7.84    25   -64    30   -31          u=4     $ Steel
24    4  -1.      26   -25    30   -31          u=4     $ Water
25    7  -2.7     27   -26    30   -31          u=4     $ Al clad
26    6  -2.66    28   -27    30   -31          u=4     $ Boral
27    7  -2.7     29   -28    30   -31          u=4     $ Al clad
28    4  -1.     129   -29    30   -31          u=4     $ water
29    5  -7.84   129   -64    32   -30          u=4     $ Steel ends
30    5  -7.84   129   -64    31   -33          u=4     $ Steel ends
31    4  -1.     129   -64    66   -32          u=4     $ Water
c
c    Type A box - Boral only on left side
c
32    0  -10   11  -12   13         u=6 fill=1 (0.8814  0.8814  0)
33    3  -6.55    60   -61    62   -63  #8      u=6     $ Zr flow channel
101   5  -7.84   74 -75 76 -77 (-70:71:-72:73)  u=6     $ DFC
```

```
34    4  -1.      64   -65    66  -118  #8 #9 #101 u=6  $ water
35    5  -7.84    118:-129:65:-66              u=6     $ Steel
36    4  -1.     129   -64    67   -118         u=6     $ Water
c
37    5  -7.84    25   -64    30   -31          u=6     $ Steel
38    4  -1.      26   -25    30   -31          u=6     $ Water
39    7  -2.7     27   -26    30   -31          u=6     $ Al clad
40    6  -2.66    28   -27    30   -31          u=6     $ Boral
41    7  -2.7     29   -28    30   -31          u=6     $ Al clad
42    4  -1.     129   -29    30   -31          u=6     $ water
43    4  -1.     129   -64    33   -67          u=6     $ Water
44    5  -7.84   129   -64    32   -30          u=6     $ Steel ends
45    5  -7.84   129   -64    31   -33          u=6     $ Steel ends
46    4  -1.     129   -64    66   -32          u=6     $ Water
c
c    Type B box -  Boral on Top only
c
47    0  -10   11  -12   13         u=7 fill=1 (0.8814  0.8814  0)
48    3  -6.55    60   -61    62   -63  #8      u=7     $ Zr flow channel
102   5  -7.84   74 -75 76 -77 (-70:71:-72:73)  u=7     $ DFC
49    4  -1.      64   -65    66   -67  #8 #9 #102 u=7  $ water
50    5  -7.84    20   -23    67   -14          u=7     $ 0.075" STEEL
51    4  -1.      20   -23    14   -15          u=7     $ WATER POCKET
52    7  -2.7     20   -23    15   -16          u=7     $ Al CLAD
53    6  -2.66    20   -23    16   -17          u=7     $ BORAL Absorber
54    7  -2.7     20   -23    17   -18          u=7     $ water
55    4  -1.      20   -23    18  -118          u=7     $ Water
56    5  -7.84    118:-129:65:-66              u=7     $ Steel
57    4  -1.      64   -21    67   -118         u=7     $ Water
58    4  -1.      24   -65    67   -118         u=7     $ water
59    5  -7.84    21   -20    67   -118         u=7     $ Steel
60    5  -7.84    23   -24    67   -118         u=7     $ Steel
61    4  -1.     129   -64    66   -118         u=7     $ Water
c
c    Type E box  - No Boral Panels
c
62    0  -10   11  -12   13         u=8 fill=1 (0.8814  0.8814  0)
63    3  -6.55    60   -61    62   -63  #8      u=8     $ Zr flow channel
103   5  -7.84   74 -75 76 -77 (-70:71:-72:73)  u=8     $ DFC
64    4  -1.     129   -65    66  -118  #8 #9 #103 u=8  $ water
65    5  -7.84    118:-129:65:-66              u=8     $ Steel
c
c    Type F box  - No Boral Panels or fuel
c
66    4  -1.     129   -65    66  -118          u=9     $ water
67    5  -7.84    118:-129:65:-66              u=9     $ Steel
c
68    4  -1.0    -34  35  -36  37  u=5  lat=1  fill=-7:6  -7:6  0:0
               5 5 5 5 5 5 5 5 5 5 5 5 5 5
               5 9 9 9 9 9 9 9 9 9 9 9 9 5
               5 9 9 9 9 9 7 4 9 9 9 9 9 5
               5 9 9 9 7 4 4 4 4 4 9 9 9 5
               5 9 9 7 4 4 4 4 4 4 4 9 9 5
               5 9 9 7 4 4 4 4 4 4 4 9 9 5
               5 9 7 4 4 4 4 4 4 4 4 4 9 5
```

```
              5 9 8 4 4 4 4 4 4 4 4 6 9 5
              5 9 9 7 4 4 4 4 4 4 4 9 9 5
              5 9 9 8 4 4 4 4 4 4 6 9 9 5
              5 9 9 9 8 4 4 4 6 6 9 9 9 5
              5 9 9 9 9 9 8 6 9 9 9 9 9 5
              5 9 9 9 9 9 9 9 9 9 9 9 9 5
              5 5 5 5 5 5 5 5 5 5 5 5 5 5
69    0           -41              50  -49  fill=5 (8.1661  8.1661  0)
70    4  -1.0    -41       43  -50            $ Water below Fuel
71    4  -1.0    -41       49  -44            $ Water above Fuel
72    5  -7.84   -42       68  -43            $ Steel below Fuel
73    5  -7.84   -42       44  -69            $ Steel above Fuel
74    5  -7.84    41  -42  43  -44            $ Radial Steel
75    0            42 :-68:  69               $ outside world

1     cz          0.6274      $ Fuel OD
2     cz          0.6280      $ Clad ID
3     cz          0.7169      $ Clad OD
4     cz          0.6280      $ Thimble ID
5     cz          0.7169      $ Thimble OD
6     px          0.8814      $ Pin Pitch
7     px         -0.8814
8     py          0.8814
9     py         -0.8814
10    px          5.4483      $ Channel ID
11    px         -5.4483
12    py          5.4483
13    py         -5.4483
14    py          7.8016
15    py          7.8155
16    py          7.8410
17    py          8.0467
18    py          8.0721
118   py          8.0861
20    px         -6.0325
21    px         -6.2230
23    px          6.0325
24    px          6.2230
25    px         -7.8016
26    px         -7.8155
27    px         -7.8410
28    px         -8.0467
29    px         -8.0721
129   px         -8.0861
30    py         -6.0325
31    py          6.0325
32    py         -6.2230
33    py          6.2230
34    px          7.6111
35    px         -8.7211
36    py          8.7211
37    py         -7.6111
41    cz          85.57
42    cz          108.43
43    pz          11.46
```

```
44    pz         331.0
49    pz         309.4         $ Top of Active F
50    pz          30.          $ Start of Active
60    px         -5.6007       $ Channel OD
61    px          5.6007
62    py         -5.6007
63    py          5.6007
64    px         -7.6111       $ Cell Box ID
65    px          7.6111
66    py         -7.6111
67    py          7.6111
68    pz         -10.13
69    pz          370.36
70    px         -6.2611       $ DFC ID
71    px          6.2611
72    py         -6.2611
73    py          6.2611
74    px         -6.5659       $ DFC OD
75    px          6.5659
76    py         -6.5659
77    py          6.5659

imp:n     1 77r 0
kcode    10000  0.94   20  120
c
sdef  par=1  erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1 -2 1.2895
c
si3  h 30. 309.
sp3  0 1
c
c
si4  s               15 16
               13 14 15 16 17 18
            12 13 14 15 16 17 18 19
            12 13 14 15 16 17 18 19
         11 12 13 14 15 16 17 18 19 20
         11 12 13 14 15 16 17 18 19 20
            12 13 14 15 16 17 18 19
            12 13 14 15 16 17 18 19
               13 14 15 16 17 18
                     15 16
sp4  1 67r
c
ds5  s               30 30
               29 29 29 29 29 29
            28 28 28 28 28 28 28 28
            27 27 27 27 27 27 27 27
         26 26 26 26 26 26 26 26 26 26
         25 25 25 25 25 25 25 25 25 25
            24 24 24 24 24 24 24 24
            23 23 23 23 23 23 23 23
               22 22 22 22 22 22
                     21 21
```

```
c
si11  -80.6831 -67.6783
si12  -64.1985 -51.1937
si13  -47.7139 -34.7091
si14  -31.2293 -18.2245
si15  -14.7447 -1.7399
si16   1.7399   14.7447
si17   18.2245  31.2293
si18   34.7091  47.7139
si19   51.1937  64.1985
si20   67.6783  80.6831
c
si21  -80.6831 -67.6783
si22  -64.1985 -51.1937
si23  -47.7139 -34.7091
si24  -31.2293 -18.2245
si25  -14.7447 -1.7399
si26   1.7399   14.7447
si27   18.2245  31.2293
si28   34.7091  47.7139
si29   51.1937  64.1985
si30   67.6783  80.6831
sp11  0 1
sp12  0 1
sp13  0 1
sp14  0 1
sp15  0 1
sp16  0 1
sp17  0 1
sp18  0 1
sp19  0 1
sp20  0 1
sp21  0 1
sp22  0 1
sp23  0 1
sp24  0 1
sp25  0 1
sp26  0 1
sp27  0 1
sp28  0 1
sp29  0 1
sp30  0 1
c
m1          92235.50c   -0.02644      $ 3.00% E Fuel
            92238.50c   -0.85504
            8016.50c    -0.11852
m3          40000.56c   1.           $ Zr Clad
m4          1001.50c    0.6667       $ Water
            8016.50c    0.3333
m5          24000.50c   0.01761      $ Steel
            25055.50c   0.001761
            26000.55c   0.05977
            28000.50c   0.008239
m6          5010.50c  1.9592E-03      $ Boral 0.0067 gm/cm2
            5011.50c  8.1175E-03
```

```
             6000.50c  2.5176E-03
             13027.50c 5.4933E-02
m7           13027.50c    1.          $ Al Clad
mt4         lwtr.01t
prdmp    j   -120   j   2
fm4      1000   1   -6
f4:n   1
sd4      1000
e4    1.000E-11   1.000E-10   5.000E-10   7.500E-10   1.000E-09   1.200E-09
      1.500E-09   2.000E-09   2.500E-09   3.000E-09
      4.700E-09   5.000E-09   7.500E-09   1.000E-08   2.530E-08
      3.000E-08   4.000E-08   5.000E-08   6.000E-08   7.000E-08
      8.000E-08   9.000E-08   1.000E-07   1.250E-07   1.500E-07
      1.750E-07   2.000E-07   2.250E-07   2.500E-07   2.750E-07
      3.000E-07   3.250E-07   3.500E-07   3.750E-07   4.000E-07
      4.500E-07   5.000E-07   5.500E-07   6.000E-07   6.250E-07
      6.500E-07   7.000E-07   7.500E-07   8.000E-07   8.500E-07
      9.000E-07   9.250E-07   9.500E-07   9.750E-07   1.000E-06
      1.010E-06   1.020E-06   1.030E-06   1.040E-06   1.050E-06
      1.060E-06   1.070E-06   1.080E-06   1.090E-06   1.100E-06
      1.110E-06   1.120E-06   1.130E-06   1.140E-06   1.150E-06
      1.175E-06   1.200E-06   1.225E-06   1.250E-06   1.300E-06
      1.350E-06   1.400E-06   1.450E-06   1.500E-06   1.590E-06
      1.680E-06   1.770E-06   1.860E-06   1.940E-06   2.000E-06
      2.120E-06   2.210E-06   2.300E-06   2.380E-06   2.470E-06
      2.570E-06   2.670E-06   2.770E-06   2.870E-06   2.970E-06
      3.000E-06   3.050E-06   3.150E-06   3.500E-06   3.730E-06
      4.000E-06   4.750E-06   5.000E-06   5.400E-06   6.000E-06
      6.250E-06   6.500E-06   6.750E-06   7.000E-06   7.150E-06
      8.100E-06   9.100E-06   1.000E-05   1.150E-05   1.190E-05
      1.290E-05   1.375E-05   1.440E-05   1.510E-05   1.600E-05
      1.700E-05   1.850E-05   1.900E-05   2.000E-05   2.100E-05
      2.250E-05   2.500E-05   2.750E-05   3.000E-05   3.125E-05
      3.175E-05   3.325E-05   3.375E-05   3.460E-05   3.550E-05
      3.700E-05   3.800E-05   3.910E-05   3.960E-05   4.100E-05
      4.240E-05   4.400E-05   4.520E-05   4.700E-05   4.830E-05
      4.920E-05   5.060E-05   5.200E-05   5.340E-05   5.900E-05
      6.100E-05   6.500E-05   6.750E-05   7.200E-05   7.600E-05
      8.000E-05   8.200E-05   9.000E-05   1.000E-04   1.080E-04
      1.150E-04   1.190E-04   1.220E-04   1.860E-04   1.925E-04
      2.075E-04   2.100E-04   2.400E-04   2.850E-04   3.050E-04
      5.500E-04   6.700E-04   6.830E-04   9.500E-04   1.150E-03
      1.500E-03   1.550E-03   1.800E-03   2.200E-03   2.290E-03
      2.580E-03   3.000E-03   3.740E-03   3.900E-03   6.000E-03
      8.030E-03   9.500E-03   1.300E-02   1.700E-02   2.500E-02
      3.000E-02   4.500E-02   5.000E-02   5.200E-02   6.000E-02
      7.300E-02   7.500E-02   8.200E-02   8.500E-02   1.000E-01
      1.283E-01   1.500E-01   2.000E-01   2.700E-01   3.300E-01
      4.000E-01   4.200E-01   4.400E-01   4.700E-01   4.995E-01
      5.500E-01   5.730E-01   6.000E-01   6.700E-01   6.790E-01
      7.500E-01   8.200E-01   8.611E-01   8.750E-01   9.000E-01
      9.200E-01   1.010E+00   1.100E+00   1.200E+00   1.250E+00
      1.317E+00   1.356E+00   1.400E+00   1.500E+00   1.850E+00
      2.354E+00   2.479E+00   3.000E+00   4.304E+00   4.800E+00
      6.434E+00   8.187E+00   1.000E+01   1.284E+01   1.384E+01
```

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1.455E+01   1.568E+01   1.733E+01   2.000E+01
```

```
HI-STAR containing MPC68F, 07x07 in DFC with 13 missing rods
c     3.00 % uniform enrichment, unreflected cask, 0.0067 g/cmsq B-10 in Boral
c
c
1     1 -10.522     -1  u=2       $ fuel
2     4 -1.0     1  -2  u=2       $ gap
3     3 -6.55    2  -3  u=2       $ Zr Clad
4     4 -1.0         3  u=2       $ water in fuel region
5     4 -1.0   -4:5     u=3       $ water in guide tubes
6     3 -6.55   4 -5    u=3       $ guide tubes
7     4 -1.0   -6   +7   -8   +9   u=1  lat=1
      fill= -4:4    -4:4   0:0
      1  1 1 1 1 1 1 1  1
      1  2 2 2 2 2 2 2  1
      1  2 1 2 1 2 1 2  1
      1  2 2 1 2 1 2 2  1
      1  2 1 2 1 2 1 2  1
      1  2 2 1 2 1 2 2  1
      1  2 1 2 1 2 1 2  1
      1  2 2 2 2 2 2 2  1
      1  1 1 1 1 1 1 1  1
c
C  BOX TYPE R
c
8     0  -10   11  -12   13         u=4 fill=1
9     3  -6.55    60   -61    62   -63  #8      u=4     $ Zr flow channel
100   5  -7.84   74 -75 76 -77 (-70:71:-72:73)  u=4     $ DFC
10    4  -1.      64   -65    66   -67  #8 #9 #100 u=4  $ water
11    5  -7.84    20   -23    67   -14          u=4     $ 0.075" STEEL
12    4  -1.      20   -23    14   -15          u=4     $ WATER POCKET
13    7  -2.7     20   -23    15   -16          u=4     $ Al CLAD
14    6  -2.66    20   -23    16   -17          u=4     $ BORAL Absorber
15    7  -2.7     20   -23    17   -18          u=4     $ Al Clad
16    4  -1.      20   -23    18  -118          u=4     $ Water
17    5  -7.84    118:-129:65:-66               u=4     $ Steel
18    4  -1.      64   -21    67   -118         u=4     $ Water
19    4  -1.      24   -65    67   -118         u=4     $ water
20    5  -7.84    21   -20    67   -118         u=4     $ Steel
21    5  -7.84    23   -24    67   -118         u=4     $ Steel
22    4  -1.     129   -64    33   -118         u=4     $ Water
c
23    5  -7.84    25   -64    30   -31          u=4     $ Steel
24    4  -1.      26   -25    30   -31          u=4     $ Water
25    7  -2.7     27   -26    30   -31          u=4     $ Al clad
26    6  -2.66    28   -27    30   -31          u=4     $ Boral
27    7  -2.7     29   -28    30   -31          u=4     $ Al clad
28    4  -1.     129   -29    30   -31          u=4     $ water
29    5  -7.84   129   -64    32   -30          u=4     $ Steel ends
30    5  -7.84   129   -64    31   -33          u=4     $ Steel ends
31    4  -1.     129   -64    66   -32          u=4     $ Water
c
c    Type A box - Boral only on left side
c
32    0  -10   11  -12   13         u=6 fill=1
33    3  -6.55    60   -61    62   -63  #8      u=6     $ Zr flow channel
```

```
101   5  -7.84   74 -75 76 -77 (-70:71:-72:73)  u=6     $ DFC
34    4  -1.      64   -65    66   -118 #8 #9 #101 u=6  $ water
35    5  -7.84    118:-129:65:-66               u=6     $ Steel
36    4  -1.     129   -64    67   -118         u=6     $ Water
c
37    5  -7.84    25   -64    30   -31          u=6     $ Steel
38    4  -1.      26   -25    30   -31          u=6     $ Water
39    7  -2.7     27   -26    30   -31          u=6     $ Al clad
40    6  -2.66    28   -27    30   -31          u=6     $ Boral
41    7  -2.7     29   -28    30   -31          u=6     $ Al clad
42    4  -1.     129   -29    30   -31          u=6     $ water
43    4  -1.     129   -64    33   -67          u=6     $ Water
44    5  -7.84   129   -64    32   -30          u=6     $ Steel ends
45    5  -7.84   129   -64    31   -33          u=6     $ Steel ends
46    4  -1.     129   -64    66   -32          u=6     $ Water
c
c    Type B box -  Boral on Top only
c
47    0  -10   11  -12   13         u=7 fill=1
48    3  -6.55    60   -61    62   -63  #8      u=7     $ Zr flow channel
102   5  -7.84   74 -75 76 -77 (-70:71:-72:73)  u=7     $ DFC
49    4  -1.      64   -65    66   -67  #8 #9 #102 u=7  $ water
50    5  -7.84    20   -23    67   -14          u=7     $ 0.075" STEEL
51    4  -1.      20   -23    14   -15          u=7     $ WATER POCKET
52    7  -2.7     20   -23    15   -16          u=7     $ Al CLAD
53    6  -2.66    20   -23    16   -17          u=7     $ BORAL Absorber
54    7  -2.7     20   -23    17   -18          u=7     $ water
55    4  -1.      20   -23    18  -118          u=7     $ Water
56    5  -7.84    118:-129:65:-66               u=7     $ Steel
57    4  -1.      64   -21    67   -118         u=7     $ Water
58    4  -1.      24   -65    67   -118         u=7     $ water
59    5  -7.84    21   -20    67   -118         u=7     $ Steel
60    5  -7.84    23   -24    67   -118         u=7     $ Steel
61    4  -1.     129   -64    66   -118         u=7     $ Water
c
c    Type E box  - No Boral Panels
c
62    0  -10   11  -12   13         u=8 fill=1
63    3  -6.55    60   -61    62   -63  #8      u=8     $ Zr flow channel
103   5  -7.84   74 -75 76 -77 (-70:71:-72:73)  u=8     $ DFC
64    4  -1.     129   -65    66  -118  #8 #9 #103 u=8  $ water
65    5  -7.84    118:-129:65:-66               u=8     $ Steel
c
c    Type F box  - No Boral Panels or fuel
c
66    4  -1.     129   -65    66  -118          u=9     $ water
67    5  -7.84    118:-129:65:-66               u=9     $ Steel
c
68    4  -1.0    -34  35  -36  37  u=5  lat=1  fill=-7:6  -7:6  0:0
               5 5 5 5 5 5 5 5 5 5 5 5 5 5
               5 9 9 9 9 9 9 9 9 9 9 9 9 5
               5 9 9 9 9 9 7 4 9 9 9 9 9 5
               5 9 9 9 7 4 4 4 4 4 9 9 9 5
               5 9 9 7 4 4 4 4 4 4 4 9 9 5
               5 9 9 7 4 4 4 4 4 4 4 9 9 5
```

```
              5 9 7 4 4 4 4 4 4 4 4 4 9 5
              5 9 8 4 4 4 4 4 4 4 4 6 9 5
              5 9 9 7 4 4 4 4 4 4 4 9 9 5
              5 9 9 8 4 4 4 4 4 4 6 9 9 5
              5 9 9 9 8 4 4 4 6 6 9 9 9 5
              5 9 9 9 9 9 8 6 9 9 9 9 9 5
              5 9 9 9 9 9 9 9 9 9 9 9 9 5
              5 5 5 5 5 5 5 5 5 5 5 5 5 5
69    0          -41             50  -49  fill=5 (8.1661  8.1661  0)
70    4  -1.0    -41       43  -50          $ Water below Fuel
71    4  -1.0    -41       49  -44          $ Water above Fuel
72    5  -7.84   -42       68  -43          $ Steel below Fuel
73    5  -7.84   -42       44  -69          $ Steel above Fuel
74    5  -7.84    41  -42  43  -44          $ Radial Steel
75    0            42 :-68:  69             $ outside world

1     cz          0.5220     $ Fuel OD
2     cz          0.5334     $ Clad ID
3     cz          0.6172     $ Clad OD
4     cz          0.5398     $ Thimble ID
5     cz          0.6261     $ Thimble OD
6     px          0.8014     $ Pin Pitch
7     px         -0.8014
8     py          0.8014
9     py         -0.8014
10    px          5.7684     $ Channel ID
11    px         -5.7684
12    py          5.7684
13    py         -5.7684
14    py          7.8016
15    py          7.8155
16    py          7.8410
17    py          8.0467
18    py          8.0721
118   py          8.0861
20    px         -6.0325
21    px         -6.2230
23    px          6.0325
24    px          6.2230
25    px         -7.8016
26    px         -7.8155
27    px         -7.8410
28    px         -8.0467
29    px         -8.0721
129   px         -8.0861
30    py         -6.0325
31    py          6.0325
32    py         -6.2230
33    py          6.2230
34    px          7.6111
35    px         -8.7211
36    py          8.7211
37    py         -7.6111
41    cz          85.57
42    cz          108.43
```

```
43    pz         11.46
44    pz         252.15
49    pz         230.66      $ Top of Active Fuel
50    pz         30.         $ Start of Active Fuel
60    px         -5.9207     $ Channel OD
61    px          5.9207
62    py         -5.9207
63    py          5.9207
64    px         -7.6111     $ Cell Box ID
65    px          7.6111
66    py         -7.6111
67    py          7.6111
68    pz         -10.13
69    pz          291.52
70    px         -6.2611       $ DFC ID
71    px          6.2611
72    py         -6.2611
73    py          6.2611
74    px         -6.5659       $ DFC OD
75    px          6.5659
76    py         -6.5659
77    py          6.5659

imp:n     1 77r 0
kcode   10000  0.94   20  120
c
sdef  par=1  erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
sp1 -2 1.2895
c
si3  h 30. 230.66
sp3  0 1
c
c
si4  s               15 16
               13 14 15 16 17 18
            12 13 14 15 16 17 18 19
            12 13 14 15 16 17 18 19
         11 12 13 14 15 16 17 18 19 20
         11 12 13 14 15 16 17 18 19 20
            12 13 14 15 16 17 18 19
            12 13 14 15 16 17 18 19
               13 14 15 16 17 18
                     15 16
sp4  1 67r
c
ds5  s               30 30
               29 29 29 29 29 29
            28 28 28 28 28 28 28 28
            27 27 27 27 27 27 27 27
         26 26 26 26 26 26 26 26 26 26
         25 25 25 25 25 25 25 25 25 25
            24 24 24 24 24 24 24 24
            23 23 23 23 23 23 23 23
               22 22 22 22 22 22
```

```
                     21 21
c
si11  -80.6831 -67.6783
si12  -64.1985 -51.1937
si13  -47.7139 -34.7091
si14  -31.2293 -18.2245
si15  -14.7447 -1.7399
si16   1.7399   14.7447
si17   18.2245  31.2293
si18   34.7091  47.7139
si19   51.1937  64.1985
si20   67.6783  80.6831
c
si21  -80.6831 -67.6783
si22  -64.1985 -51.1937
si23  -47.7139 -34.7091
si24  -31.2293 -18.2245
si25  -14.7447 -1.7399
si26   1.7399   14.7447
si27   18.2245  31.2293
si28   34.7091  47.7139
si29   51.1937  64.1985
si30   67.6783  80.6831
sp11  0 1
sp12  0 1
sp13  0 1
sp14  0 1
sp15  0 1
sp16  0 1
sp17  0 1
sp18  0 1
sp19  0 1
sp20  0 1
sp21  0 1
sp22  0 1
sp23  0 1
sp24  0 1
sp25  0 1
sp26  0 1
sp27  0 1
sp28  0 1
sp29  0 1
sp30  0 1
c
m1           92235.50c   -0.02644      $ 3.00% E Fuel
             92238.50c   -0.85504
             8016.50c    -0.11852
m3           40000.56c   1.           $ Zr Clad
m4           1001.50c    0.6667       $ Water
             8016.50c    0.3333
m5           24000.50c   0.01761      $ Steel
             25055.50c   0.001761
             26000.55c   0.05977
             28000.50c   0.008239
m6           5010.50c  1.9592E-03      $ Boral 0.0067 gm/cm2
```

```
             5011.50c  8.1175E-03
             6000.50c  2.5176E-03
             13027.50c 5.4933E-02
m7           13027.50c    1.          $ Al Clad
mt4         lwtr.01t
prdmp    j   -120   j   2
fm4      1000   1   -6
f4:n   1
sd4      1000
e4    1.000E-11   1.000E-10   5.000E-10   7.500E-10   1.000E-09   1.200E-09
      1.500E-09   2.000E-09   2.500E-09   3.000E-09
      4.700E-09   5.000E-09   7.500E-09   1.000E-08   2.530E-08
      3.000E-08   4.000E-08   5.000E-08   6.000E-08   7.000E-08
      8.000E-08   9.000E-08   1.000E-07   1.250E-07   1.500E-07
      1.750E-07   2.000E-07   2.250E-07   2.500E-07   2.750E-07
      3.000E-07   3.250E-07   3.500E-07   3.750E-07   4.000E-07
      4.500E-07   5.000E-07   5.500E-07   6.000E-07   6.250E-07
      6.500E-07   7.000E-07   7.500E-07   8.000E-07   8.500E-07
      9.000E-07   9.250E-07   9.500E-07   9.750E-07   1.000E-06
      1.010E-06   1.020E-06   1.030E-06   1.040E-06   1.050E-06
      1.060E-06   1.070E-06   1.080E-06   1.090E-06   1.100E-06
      1.110E-06   1.120E-06   1.130E-06   1.140E-06   1.150E-06
      1.175E-06   1.200E-06   1.225E-06   1.250E-06   1.300E-06
      1.350E-06   1.400E-06   1.450E-06   1.500E-06   1.590E-06
      1.680E-06   1.770E-06   1.860E-06   1.940E-06   2.000E-06
      2.120E-06   2.210E-06   2.300E-06   2.380E-06   2.470E-06
      2.570E-06   2.670E-06   2.770E-06   2.870E-06   2.970E-06
      3.000E-06   3.050E-06   3.150E-06   3.500E-06   3.730E-06
      4.000E-06   4.750E-06   5.000E-06   5.400E-06   6.000E-06
      6.250E-06   6.500E-06   6.750E-06   7.000E-06   7.150E-06
      8.100E-06   9.100E-06   1.000E-05   1.150E-05   1.190E-05
      1.290E-05   1.375E-05   1.440E-05   1.510E-05   1.600E-05
      1.700E-05   1.850E-05   1.900E-05   2.000E-05   2.100E-05
      2.250E-05   2.500E-05   2.750E-05   3.000E-05   3.125E-05
      3.175E-05   3.325E-05   3.375E-05   3.460E-05   3.550E-05
      3.700E-05   3.800E-05   3.910E-05   3.960E-05   4.100E-05
      4.240E-05   4.400E-05   4.520E-05   4.700E-05   4.830E-05
      4.920E-05   5.060E-05   5.200E-05   5.340E-05   5.900E-05
      6.100E-05   6.500E-05   6.750E-05   7.200E-05   7.600E-05
      8.000E-05   8.200E-05   9.000E-05   1.000E-04   1.080E-04
      1.150E-04   1.190E-04   1.220E-04   1.860E-04   1.925E-04
      2.075E-04   2.100E-04   2.400E-04   2.850E-04   3.050E-04
      5.500E-04   6.700E-04   6.830E-04   9.500E-04   1.150E-03
      1.500E-03   1.550E-03   1.800E-03   2.200E-03   2.290E-03
      2.580E-03   3.000E-03   3.740E-03   3.900E-03   6.000E-03
      8.030E-03   9.500E-03   1.300E-02   1.700E-02   2.500E-02
      3.000E-02   4.500E-02   5.000E-02   5.200E-02   6.000E-02
      7.300E-02   7.500E-02   8.200E-02   8.500E-02   1.000E-01
      1.283E-01   1.500E-01   2.000E-01   2.700E-01   3.300E-01
      4.000E-01   4.200E-01   4.400E-01   4.700E-01   4.995E-01
      5.500E-01   5.730E-01   6.000E-01   6.700E-01   6.790E-01
      7.500E-01   8.200E-01   8.611E-01   8.750E-01   9.000E-01
      9.200E-01   1.010E+00   1.100E+00   1.200E+00   1.250E+00
      1.317E+00   1.356E+00   1.400E+00   1.500E+00   1.850E+00
      2.354E+00   2.479E+00   3.000E+00   4.304E+00   4.800E+00
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6.434E+00   8.187E+00   1.000E+01   1.284E+01   1.384E+01
1.455E+01   1.568E+01   1.733E+01   2.000E+01
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=NITAWL
' k4rf5f45, HI-STAR containing MPC24, 15x15 assembly a17 @ 4.1% E
0$$ 84 E
1$$ 0 13  0  4R0  1  E T
2$$ 92235    92238
    40000    1001     8016     5010      5011      6012
    13027    24000    25055     26000     28000
3** 92238 294.6    2   .4752    .1928     0.      0.02251     1
   16.0   7.8209   1   235.04  0.5154    1      1.0           T
END
=KENO5A
k4rf5f45, HI-STAR containing MPC24, 15x15 assembly a17 @ 4.1% E
READ PARAM TME=10000 GEN=300 NPG=10000 NSK=50 LIB=4 TBA=5
END PARAM
READ MIXT SCT=2  EPS=1.0
MIX=1   92235  9.7463E-04
        92238  0.02251
        8016   0.04694
MIX=2   40000  0.04323
MIX=3   1001   0.06688
        8016   0.03344
MIX=4   24000  0.01761
        25055  0.001761
        26000  0.05977
        28000  0.008239
MIX=5   5010   8.7066E-03
        5011   3.5116E-02
        6012   1.0948E-02
        13027  3.6936E-02
MIX=6   1001   0.06688
        8016   0.03344
MIX=7   13027  0.06026
END MIXT
'
' cell-id      8.98
' cell-pitch   10.906
' wall-thkns   5/16
' angle-thkns  5/16
' boral-gap    0.0035
' boral-pocket 0.082
' boral-thkns  0.075
' boral-clad   0.01
' boral-core   0.055
' sheathing    0.0235
' boral-wide   7.500
' boral-narrow 6.250
'
' gap size     1.09
'
READ GEOM
UNIT           1
COM="FUEL ROD"
CYLINDER        1  1  0.4752                                   381.0   0.
CYLINDER        3  1  0.4851                                   381.0   0.
CYLINDER        2  1  0.5436                                   381.0   0.
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CUBOID          3  1  0.7214   -0.7214    0.7214    -0.7214     381.0   0.
UNIT           2
COM="GUIDE  TUBE CELL"
CYLINDER        3  1  0.6350                                    381.0   0.
CYLINDER        2  1  0.6706                                    381.0   0.
CUBOID          3  1  0.7214   -0.7214    0.7214    -0.7214     381.0   0.
UNIT           4
COM="LONG HORIZONTAL BORAL PANEL - NORTH"
CUBOID          5  1  9.525  -9.525   0.06985    -0.06985     381.0   0.
CUBOID          7  1  9.525  -9.525   0.09525    -0.09525     381.0   0.
CUBOID          3  1  9.525  -9.525   0.10414    -0.10414     381.0   0.
CUBOID          4  1  9.58469  -9.58469   0.16383      -0.10414     381.0   0.
UNIT           5
COM="LONG VERTICAL BORAL PANEL - EAST"
CUBOID          5  1  0.06985    -0.06985    9.525  -9.525    381.0   0.
CUBOID          7  1  0.09525    -0.09525    9.525  -9.525    381.0   0.
CUBOID          3  1  0.10414    -0.10414    9.525  -9.525    381.0   0.
CUBOID          4  1  0.16383      -0.10414    9.58469  -9.58469    381.0   0.
UNIT           6
COM="LONG HORIZONTAL BORAL PANEL - SOUTH"
CUBOID          5  1  9.525  -9.525    0.06985   -0.06985     381.0   0.
CUBOID          7  1  9.525  -9.525    0.09525   -0.09525     381.0   0.
CUBOID          3  1  9.525  -9.525    0.10414   -0.10414     381.0   0.
CUBOID          4  1  9.58469  -9.58469    0.10414   -0.16383       381.0   0.
UNIT           7
COM="LONG VERTICAL BORAL PANEL - WEST"
CUBOID          5  1  0.06985    -0.06985    9.525  -9.525    381.0   0.
CUBOID          7  1  0.09525    -0.09525    9.525  -9.525    381.0   0.
CUBOID          3  1  0.10414    -0.10414    9.525  -9.525    381.0   0.
CUBOID          4  1  0.10414    -0.16383      9.58469  -9.58469    381.0   0.
UNIT           8  ARRAY 1  -10.8204    -10.8204       0.
COM="CENTRAL FUEL ASSEMBLIES - 4 BORAL PANELS"
CUBOID          3  1  11.4046  -11.4046   11.4046  -11.4046     381.0   0.
CUBOID          4  1  12.1984  -12.1984   12.1984  -12.1984     381.0   0.
CUBOID          3  1  12.4673  -12.4673   12.4673  -12.4673     381.0   0.
HOLE            4    0.          12.3026     0.
HOLE            5   12.3026       0.         0.
HOLE            6  0.          -12.3026    0.
HOLE            7  -12.3026      0.        0.
UNIT          21  ARRAY 1  -10.8204    -10.8204       0.
COM="CENTRAL FUEL ASSEMBLIES - 4 BORAL PANELS, W/O EAST WALL"
CUBOID          3  1  11.4046  -11.4046   11.4046  -11.4046     381.0   0.
CUBOID          4  1  11.4046  -12.1984   12.1984  -12.1984     381.0   0.
CUBOID          3  1  11.4046  -12.4673   12.4673  -12.4673     381.0   0.
HOLE            4    0.          12.3026     0.
HOLE            6  0.          -12.3026    0.
HOLE            7  -12.3026      0.        0.
UNIT          22  ARRAY 1  -10.8204    -10.8204       0.
COM="CENTRAL FUEL ASSEMBLIES - 4 BORAL PANELS, W/O WEST WALL"
CUBOID          3  1  11.4046  -11.4046   11.4046  -11.4046     381.0   0.
CUBOID          4  1  12.1984  -11.4046   12.1984  -12.1984     381.0   0.
CUBOID          3  1  12.4673  -11.4046   12.4673  -12.4673     381.0   0.
HOLE            4    0.          12.3026     0.
HOLE            5   12.3026       0.         0.
HOLE            6  0.          -12.3026    0.
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UNIT          23
COM="CELL WALL BETWEEN UNITS 21 AND 22"
CUBOID          4  1  0.396775  -0.396775   23.9998  -23.9998     381.0   0.
UNIT           9
COM="SHORT HORIZONTAL BORAL PANEL - NORTH"
CUBOID          5  1  7.9375  -7.9375   0.06985    -0.06985     381.0   0.
CUBOID          7  1  7.9375  -7.9375   0.09525    -0.09525     381.0   0.
CUBOID          3  1  7.9375  -7.9375   0.10414    -0.10414     381.0   0.
CUBOID          4  1  7.99719  -7.99719   0.16383      -0.10414     381.0   0.
UNIT          10
COM="SHORT VERTICAL BORAL PANEL - EAST"
CUBOID          5  1  0.06985    -0.06985    7.9375  -7.9375    381.0   0.
CUBOID          7  1  0.09525    -0.09525    7.9375  -7.9375    381.0   0.
CUBOID          3  1  0.10414    -0.10414    7.9375  -7.9375    381.0   0.
CUBOID          4  1  0.16383      -0.10414    7.99719  -7.99719    381.0   0.
UNIT          11
COM="SHORT HORIZONTAL BORAL PANEL - SOUTH"
CUBOID          5  1  7.9375  -7.9375    0.06985   -0.06985     381.0   0.
CUBOID          7  1  7.9375  -7.9375    0.09525   -0.09525     381.0   0.
CUBOID          3  1  7.9375  -7.9375    0.10414   -0.10414     381.0   0.
CUBOID          4  1  7.99719  -7.99719    0.10414   -0.16383       381.0   0.
UNIT          12
COM="SHORT VERTICAL BORAL PANEL - WEST"
CUBOID          5  1  0.06985    -0.06985    7.9375  -7.9375    381.0   0.
CUBOID          7  1  0.09525    -0.09525    7.9375  -7.9375    381.0   0.
CUBOID          3  1  0.10414    -0.10414    7.9375  -7.9375    381.0   0.
CUBOID          4  1  0.10414    -0.16383      7.99719  -7.99719    381.0   0.
UNIT          13  ARRAY 1  -10.8204    -10.8204       0.
COM="Array B short Boral N & E "
CUBOID          3  1  11.4046  -11.4046   11.4046  -11.4046     381.0   0.
CUBOID          4  1  12.1984  -12.1984   12.1984  -12.1984     381.0   0.
CUBOID          3  1  12.4673  -12.4673   12.4673  -12.4673     381.0   0.
HOLE            9    0.         12.3026     0.
HOLE           10   12.3026       0.        0.
HOLE            6  0.           -12.3026    0.
HOLE            7  -12.3026      0.        0.
UNIT          14  ARRAY 1  -10.8204    -10.8204       0.
COM="Array C short Boral E & S "
CUBOID          3  1  11.4046  -11.4046   11.4046  -11.4046     381.0   0.
CUBOID          4  1  12.1984  -12.1984   12.1984  -12.1984     381.0   0.
CUBOID          3  1  12.4673  -12.4673   12.4673  -12.4673     381.0   0.
HOLE            4    0.         12.3026     0.
HOLE           10   12.3026       0.        0.
HOLE           11  0.           -12.3026    0.
HOLE            7  -12.3026      0.        0.
UNIT          15  ARRAY 1  -10.8204    -10.8204       0.
COM="Array D short Boral S & W "
CUBOID          3  1  11.4046  -11.4046   11.4046  -11.4046     381.0   0.
CUBOID          4  1  12.1984  -12.1984   12.1984  -12.1984     381.0   0.
CUBOID          3  1  12.4673  -12.4673   12.4673  -12.4673     381.0   0.
HOLE            4    0.         12.3026     0.
HOLE            5   12.3026       0.        0.
HOLE           11  0.           -12.3026    0.
HOLE           12  -12.3026      0.        0.
UNIT          16  ARRAY 1  -10.8204    -10.8204       0.
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COM="Array E short Boral N & W "
CUBOID          3  1  11.4046  -11.4046   11.4046  -11.4046     381.0   0.
CUBOID          4  1  12.1984  -12.1984   12.1984  -12.1984     381.0   0.
CUBOID          3  1  12.4673  -12.4673   12.4673  -12.4673     381.0   0.
HOLE            9    0.         12.3026     0.
HOLE            5   12.3026       0.        0.
HOLE            6  0.          -12.3026    0.
HOLE           12  -12.3026      0.        0.
UNIT           17
CUBOID          4  1  0.7938   -0.        1.60      -0.         381.0   0.
UNIT           18
CUBOID          4  1  1.60     -0.        0.7938    -0.         381.0   0.
UNIT           30
CUBOID          4  1  1.37332  -1.37332   3.4925  -3.4925     381.0   0.
UNIT           31
CUBOID          4  1  1.05959  -1.05959   1.37332  -1.37332     381.0   0.
UNIT           41
CUBOID          3  1  14.6768  -14.6768   1.65227  -1.65227     381.0   0.
CUBOID          4  1  14.6768  -14.6768   2.44602  -2.44602     381.0   0.
UNIT           42
CUBOID          3  1  1.65227  -1.65227   14.6768  -14.6768     381.0   0.
CUBOID          4  1  2.44602  -2.44602   14.6768  -14.6768     381.0   0.
GLOBAL
UNIT           19
COM="ASSEMBLY ARRAY + X DIRECTION"
CYLINDER        3  1  86.57                                   396.24  -10.16
HOLE            8   13.8506    13.8506    0.
HOLE            8   13.8506   -13.8506    0.
HOLE           21   17.949    41.5519    0.
HOLE           21   17.949   -41.5519    0.
HOLE           13   13.8506    69.2531    0.
HOLE           14   13.8506   -69.2531    0.
HOLE           22   41.5519    17.949    0.
HOLE           22   41.5519   -17.949    0.
HOLE           13   45.6502    45.6502    0.
HOLE           14   45.6502   -45.6502    0.
HOLE           13   69.2531    13.8506    0.
HOLE           14   69.2531   -13.8506    0.
HOLE            8  -13.8506    13.8506    0.
HOLE            8  -13.8506   -13.8506    0.
HOLE           22  -17.949    41.5519    0.
HOLE           22  -17.949   -41.5519    0.
HOLE           16  -13.8506    69.2531    0.
HOLE           15  -13.8506   -69.2531    0.
HOLE           21  -41.5519    17.949    0.
HOLE           21  -41.5519   -17.949    0.
HOLE           16  -45.6502    45.6502    0.
HOLE           15  -45.6502   -45.6502    0.
HOLE           16  -69.2531    13.8506    0.
HOLE           15  -69.2531   -13.8506    0.
HOLE           23   29.7505    29.7505    0.
HOLE           23  -29.7505    29.7505    0.
HOLE           23   29.7505   -29.7505    0.
HOLE           23  -29.7505   -29.7505    0.
HOLE            5   30.2516    41.5519    0.
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HOLE            5   30.2516   -41.5519    0.
HOLE            7  -30.2516    41.5519    0.
HOLE            7  -30.2516   -41.5519    0.
HOLE            7   29.2494    17.949    0.
HOLE            7   29.2494   -17.949    0.
HOLE            5  -29.2494    17.949    0.
HOLE            5  -29.2494   -17.949    0.
HOLE           30      0.       0.    0.
HOLE           31   2.43291       0.    0.
HOLE           31  -2.43291       0.    0.
HOLE           41   41.5519       0.    0.
HOLE           41  -41.5519       0.    0.
HOLE           42      0.    41.5519    0.
HOLE           42      0.   -41.5519    0.
CYLINDER        4  1  108.43                                 435.61  -31.75
CUBOID          3  1  139   -139    139   -139    435.61   -31.75
END GEOM
READ ARRAY
ARA=1  NUX=15  NUY=15  NUZ=1
COM="15 X 15 FUEL ASSEMBLY"
FILL
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 1 1 1 2 1 1 1 2 1 1 1 1 1
      1 1 1 2 1 1 1 1 1 1 1 2 1 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 2 1 1 2 1 1 1 2 1 1 2 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 1 1 1 1 1 2 1 1 1 1 1 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 2 1 1 2 1 1 1 2 1 1 2 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 1 2 1 1 1 1 1 1 1 2 1 1 1
      1 1 1 1 1 2 1 1 1 2 1 1 1 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
      1 1 1 1 1 1 1 1 1 1 1 1 1 1 1
END FILL
END ARRAY
END DATA
END
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=NITAWL
' HI-STAR containing MPC68, 08x08 @ 4.2% E
0$$ 84 E
1$$ 0 13  0  4R0  1  E T
2$$ 92235    92238
    40000    1001     8016     5010      5011      6012
    13027    24000    25055     26000     28000
3** 92238 294.6    2   .5207    .1623     0.      0.02248     1
   16.0   7.8330   1   235.04  0.5662    1      1.0           T
END
=KENO5A
HI-STAR containing MPC68, 08x08 @ 4.2% E
'  GE 8X8R FUEL 2 WATER HOLES
READ PARAM TME=10000 GEN=1100 NPG=5000 NSK=100
     LIB=4 TBA=5  LNG=400000   NB8=900
END PARAM
READ MIXT SCT=2  EPS=1.
MIX=1   92235  9.983E-04
        92238  0.02248
        8016   0.04697
MIX=2   40000  0.04323
MIX=3   1001   0.06688
        8016   0.03344
MIX=4   24000  0.01761
        25055  0.001761
        26000  0.05977
        28000  0.008239
MIX=5   5010   8.071E-03
        5011   3.255E-02
        6012   1.015E-02
        13027  3.805E-02
MIX=6   13027  0.06026
END MIXT
READ GEOM
UNIT           1
COM=           "FUEL ROD"
CYLINDER        1  1  0.5207                                    381.0   0.
CYLINDER        3  1  0.5321                                    381.0   0.
CYLINDER        2  1  0.6134                                    381.0   0.
CUBOID          3  1  0.8128   -0.8128    0.8128    -0.8128     381.0   0.
UNIT           2
COM=           "GUIDE  TUBE CELL"
CYLINDER        3  1  0.6744                                    381.0   0.
CYLINDER        2  1  0.7506                                    381.0   0.
CUBOID          3  1  0.8128   -0.8128    0.8128    -0.8128     381.0   0.
UNIT           4
COM=           "HORIZONTAL BORAL PANEL"
CUBOID          5  1  6.0325   -6.0325    0.1027    -0.1027     381.0   0.
CUBOID          6  1  6.0325   -6.0325    0.1283    -0.1283     381.0   0.
CUBOID          3  1  6.0325   -6.0325    0.1422    -0.1422     381.0   0.
CUBOID          4  1  6.4611   -6.4611    0.1422    -0.3327     381.0   0.
UNIT           5
COM=           "VERTICAL BORAL PANEL"
CUBOID          5  1  0.1027   -0.1027     6.0325   -6.0325    381.0   0.
CUBOID          6  1  0.1283   -0.1283     6.0325   -6.0325    381.0   0.
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CUBOID          3  1  0.1422   -0.1422     6.0325   -6.0325    381.0   0.
CUBOID          4  1  0.3327   -0.1422     6.4611   -6.4611    381.0   0.
UNIT           8  ARRAY 1  -6.5024     -6.5024        0.
COM=           "FUEL ASSEMBLIES - 2 BORAL PANELS"
CUBOID          3  1  6.7031   -6.7031    6.7031   -6.7031          381.0   0.
CUBOID          2  1  6.9571   -6.9571    6.9571   -6.9571          381.0   0.
CUBOID          3  1  7.6111   -8.0861    8.0861   -7.6111          381.0   0.
HOLE            4    0.          7.9438     0.
HOLE            5  -7.9438       0.         0.
CUBOID          4  1  7.6111   -8.7211    8.7211   -7.6111          381.0   0.
UNIT           9  ARRAY 1  -6.5024     -6.5024        0.
COM=           "FUEL ASSEMBLIES - Type A"
CUBOID          3  1  6.7031   -6.7031    6.7031   -6.7031          381.0   0.
CUBOID          2  1  6.9571   -6.9571    6.9571   -6.9571          381.0   0.
CUBOID          3  1  7.6111   -8.0861    8.0861   -7.6111          381.0   0.
HOLE            5  -7.9438       0.         0.
CUBOID          4  1  8.2461   -8.7211    8.7211   -7.6111          381.0   0.
UNIT          10  ARRAY 1  -6.5024     -6.5024        0.
COM=           "FUEL ASSEMBLIES - Type B"
CUBOID          3  1  6.7031   -6.7031    6.7031   -6.7031          381.0   0.
CUBOID          2  1  6.9571   -6.9571    6.9571   -6.9571          381.0   0.
CUBOID          3  1  7.6111   -8.0861    8.0861   -7.6111          381.0   0.
HOLE            5  -7.9438       0.         0.
CUBOID          4  1  7.6111   -8.7211    8.7211   -7.6111          381.0   0.
UNIT          11  ARRAY 1  -6.5024     -6.5024        0.
COM=           "FUEL ASSEMBLIES - Type C"
CUBOID          3  1  6.7031   -6.7031    6.7031   -6.7031          381.0   0.
CUBOID          2  1  6.9571   -6.9571    6.9571   -6.9571          381.0   0.
CUBOID          3  1  7.6111   -8.0861    8.0861   -7.6111          381.0   0.
HOLE            4    0.          7.9438     0.
HOLE            5  -7.9438       0.         0.
CUBOID          4  1  8.2461   -8.7211    8.7211   -7.6111          381.0   0.
UNIT          12  ARRAY 1  -6.5024     -6.5024        0.
COM=           "FUEL ASSEMBLIES - Type D"
CUBOID          3  1  6.7031   -6.7031    6.7031   -6.7031          381.0   0.
CUBOID          2  1  6.9571   -6.9571    6.9571   -6.9571          381.0   0.
CUBOID          3  1  7.6111   -8.0861    8.0861   -7.6111          381.0   0.
HOLE            4    0.          7.9438     0.
HOLE            5  -7.9438       0.         0.
CUBOID          4  1  8.2461   -8.7211    8.7211   -8.2461          381.0   0.
UNIT          13  ARRAY 1  -6.5024     -6.5024        0.
COM=           "FUEL ASSEMBLIES - Type E"
CUBOID          3  1  6.7031   -6.7031    6.7031   -6.7031          381.0   0.
CUBOID          2  1  6.9571   -6.9571    6.9571   -6.9571          381.0   0.
CUBOID          3  1  7.6111   -8.0861    8.0861   -7.6111          381.0   0.
HOLE            4    0.          7.9438     0.
HOLE            5  -7.9438       0.         0.
CUBOID          4  1  7.6111   -8.7211    8.7211   -8.2461          381.0   0.
UNIT          14  ARRAY 1  -6.5024     -6.5024        0.
COM=           "FUEL ASSEMBLIES - Type F"
CUBOID          3  1  6.7031   -6.7031    6.7031   -6.7031          381.0   0.
CUBOID          2  1  6.9571   -6.9571    6.9571   -6.9571          381.0   0.
CUBOID          3  1  7.6111   -8.0861    8.0861   -7.6111          381.0   0.
HOLE            4    0.          7.9438     0.
CUBOID          4  1  7.6111   -8.7211    8.7211   -8.2461          381.0   0.
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UNIT          15  ARRAY 1  -6.5024     -6.5024        0.
COM=           "FUEL ASSEMBLIES - TYPE G"
CUBOID          3  1  6.7031   -6.7031    6.7031   -6.7031          381.0   0.
CUBOID          2  1  6.9571   -6.9571    6.9571   -6.9571          381.0   0.
CUBOID          3  1  7.6111   -8.0861    8.0861   -7.6111          381.0   0.
HOLE            4    0.         7.9438     0.
CUBOID          4  1  7.6111   -8.7211    8.7211   -7.6111          381.0   0.
UNIT          16  ARRAY 1  -6.5024     -6.5024        0.
COM=           "FUEL ASSEMBLIES - TYPE H"
CUBOID          3  1  6.7031   -6.7031    6.7031   -6.7031          381.0   0.
CUBOID          2  1  6.9571   -6.9571    6.9571   -6.9571          381.0   0.
CUBOID          3  1  7.6111   -8.0861    8.0861   -7.6111          381.0   0.
CUBOID          4  1  7.6111   -8.7211    8.7211   -7.6111          381.0   0.
UNIT           17  ARRAY 2     -48.9966  -48.9966          0
UNIT           18  ARRAY 3     -16.3322  -7.6111           0.
UNIT           19  ARRAY 4     -48.9966  -7.6111           0.
UNIT           20  ARRAY 5     -8.7211   -16.3322          0.
UNIT           21  ARRAY 6     -8.7211   -50.1068          0.
UNIT           22  ARRAY 7     -8.7211   -16.3322          0.
UNIT           23  ARRAY 8     -8.7211   -16.3322          0.
UNIT           24  ARRAY 9     -8.7211   -16.3322          0.
UNIT           25  ARRAY 10    -8.7211   -16.3322          0.
UNIT           26  ARRAY 11    -16.3322  -8.7213           0.
UNIT           27  ARRAY 12    -16.3322  -7.6111           0.
UNIT           28  ARRAY 13    -16.3322  -8.7213           0.
UNIT           29  ARRAY 14    -16.3322  -8.7213           0.
GLOBAL
UNIT           30
CYLINDER        3  1   85.57                                       402.5   -18.54.
HOLE           17      0.0        0.0        0.
HOLE           18      0.0       73.4949     0.
HOLE           19      0.0       57.1627     0.
HOLE           20     -73.4949    0.0        0.
HOLE           21     -56.6077    0.0        0.
HOLE           22      57.7177   32.6644     0.
HOLE           23      57.7177    0.0        0.
HOLE           24      74.052     0.0        0.
HOLE           25      57.7177  -32.6644     0.
HOLE           26      32.6644  -57.7177     0.
HOLE           27       0.0     -57.7177     0.
HOLE           28     -32.6644  -57.7177     0.
HOLE           29       0.0     -74.052      0.
CYLINDER        4   1   108.43                                  441.85  -40.13
CUBOID          3   1   109.     -109.      109.     -109.    442     -40.2
END GEOM
READ ARRAY
ARA=1  NUX=8   NUY=8   NUZ=1
COM=   "8 X 8 FUEL ASSEMBLY"
FILL
 1 1 1 1 1 1 1 1
 1 1 1 1 1 1 1 1
 1 1 1 1 1 1 1 1
 1 1 1 2 1 1 1 1
 1 1 1 1 2 1 1 1
 1 1 1 1 1 1 1 1
```

```
 1 1 1 1 1 1 1 1
 1 1 1 1 1 1 1 1
END FILL
ARA=2  NUX=6   NUY=6   NUZ=1
COM=   "6 X 6 CENTRAL ARRAY OF FUEL ASSEMBLIES"
FILL
  8 8 8 8 8 8
  8 8 8 8 8 8
  8 8 8 8 8 8
  8 8 8 8 8 8
  8 8 8 8 8 8
  8 8 8 8 8 8
END FILL
ARA=3  NUX=2   NUY=1   NUZ=1
COM=   "2 X 1  ARRAY OF FUEL ASSEMBLIES - TOP ROW"
FILL
 16  9
END FILL
ARA=4  NUX=6   NUY=1   NUZ=1
COM=   "6 X 1  ARRAY OF FUEL ASSEMBLIES - 2ND ROW"
FILL
  16  10  8  8  10  9
END FILL
ARA=5  NUX=1   NUY=2   NUZ=1
COM=   "2 X 1  ARRAY OF FUEL ASSEMBLIES - OUTER LEFT"
FILL
  14
  16
END FILL
ARA=6  NUX=1   NUY=6   NUZ=1
COM=   "1 X 6  ARRAY OF FUEL ASSEMBLIES - 2ND ROW LEFT"
FILL
  14  15  8  8  15  16
END FILL
ARA=7  NUX=1   NUY=2   NUZ=1
COM=   "1 X 2  ARRAY OF FUEL ASSEMBLIES - UPPER RIGHT"
FILL
  11
   9
END FILL
ARA=8  NUX=1   NUY=2   NUZ=1
COM=   "1 X 2  ARRAY OF FUEL ASSEMBLIES - MIDDLE RIGHT"
FILL
   8
   8
END FILL
ARA=9  NUX=1   NUY=2   NUZ=1
COM=   "1 X 2  ARRAY OF FUEL ASSEMBLIES - MIDDLE RIGHT"
FILL
   11
    9
END FILL
ARA=10  NUX=1   NUY=2   NUZ=1
COM=   "1 X 2  ARRAY OF FUEL ASSEMBLIES - LOWER RIGHT"
FILL
```

```
  11
  11
END FILL
ARA=11  NUX=2   NUY=1   NUZ=1
COM=   "2 X 1  ARRAY OF FUEL ASSEMBLIES - 2ND BOTTOM ROW"
FILL
  13  13
END FILL
ARA=12  NUX=2   NUY=1   NUZ=1
COM=   "2 X 1  ARRAY OF FUEL ASSEMBLIES - BOTTOM ROW"
FILL
  8  8
END FILL
ARA=13  NUX=2   NUY=1   NUZ=1
COM=   "2 X 1  ARRAY OF FUEL ASSEMBLIES - BOTTOM ROW"
FILL
  14  13
END FILL
ARA=14  NUX=2   NUY=1   NUZ=1
COM=   "2 X 1  ARRAY OF FUEL ASSEMBLIES - BOTTOM ROW"
FILL
  14 13
END FILL
END ARRAY
END DATA
END
```

CHAPTER 7*: CONFINEMENT

7.0 INTRODUCTION

Confinement of all radioactive materials in the HI-STAR 100 *system is provided by the MPC. The design of the HI-STAR 100 MPC assures that there are no credible design basis events that would result in a radiological release to the environment. The* HI-STAR 100 *overpack and HI-TRAC transfer cask are designed to provide physical protection to the MPC during normal, off-normal, and postulated accident conditions to assure that the integrity of the MPC is maintained. The dry inert atmosphere in the MPC and the passive heat removal capabilities of the HI-STAR 100 also assure that the SNF assemblies remain protected from long-term degradation.*

A detailed description of the confinement structures, systems, and components important to safety is provided in Chapter 2. The structural adequacy of the MPC is demonstrated by the analyses documented in Chapter 3. The physical protection of the MPC provided by the overpack and the HI-TRAC Transfer Cask is demonstrated by the structural analyses documented in Chapter 3 for off-normal and postulated accident conditions that are considered in Chapter 11. The heat removal capabilities of the HI-STAR 100 *system are demonstrated by the thermal analyses documented in Chapter 4. Materials evaluation in Chapter 8 demonstrates the compatibility and durability of the MPC materials for long term spent fuel storage.*

This chapter describes the HI-STAR 100 *confinement design and describes how the design satisfies the confinement requirements of 10CFR72 [7.0.1]. It also provides an evaluation of the MPC confinement boundary as it relates to the criteria contained in Interim Staff Guidance (ISG)–18 [7.0.2] and applicable portions of ANSI N14.5-1997 [7.0.3] as justification for reaching the determination that leakage from the confinement boundary is not credible and, therefore, a quantification of the consequence of leakage from the MPC is not required. This chapter is in general compliance with NUREG-1536 [7.0.4] as noted in Chapter 1.*

It should be observed that the configuration of the confinement boundary of the MPCs covered by this FSAR is identical to that used in the MPCs in Docket No. 72-1014 (HI-STORM 100 system), including weld joint details and weld types and weld sizes. Therefore, it is reasonable to conclude that the safety evaluation conducted to establish confinement integrity in Docket No. 72-1014 is also applicable herein.

*Much of the new information in this chapter is directly extracted from previously NRC approved Holtec dockets; this information is shown in italics. In Chapter 7, this information was extracted from Reference [7.0.5]. All changes in this revision are marked with revision bars. *This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536 as described in Chapter 1.*

7.1 CONFINEMENT BOUNDARY

The confinement against the release of radioactive contents is the all welded MPC. There are no bolted closures or mechanical seals in the MPC confinement boundary.

The confinement boundary of the MPC consists of the following parts:

- *MPC shell*
- *MPC bottom plate*
- *MPC lid*
- *MPC vent and drain port covers*
- *MPC closure ring*
- *associated welds*

The combination of the welded MPC lid and the welded closure ring form the redundant closure of the MPC and satisfies the requirements of 10 CFR 72.236(e) [7.0.1]. The confinement boundary is shown in the licensing drawing package in Section 1.5. Chapter 2 provides design criteria for the confinement boundary. All components of the confinement boundary are important-to-safety, as specified on the licensing drawings. The MPC confinement boundary is designed, fabricated, inspected and tested in accordance with the applicable requirements of ASME Code, Section III, Subsection NB [7.1.1], with alternatives given in Chapter 2.

7.1.1 Confinement Vessel

The HI-STAR 100 *system confinement vessel is the MPC. The MPC is designed to provide confinement of all radionuclides under normal, off-normal and accident conditions. The three major components of the MPC vessel are the shell, baseplate, and lid. The shell welds and the shell to baseplate weld are performed at the fabrication facility. The remaining confinement boundary welds are performed in the field (Table 7.1.1).*

The MPC lid consists of two sections (referred to as upper and lower) welded together. Only the upper portion of the lid is credited in the confinement boundary. The lid is made intentionally thick by the addition of the lower portion of the lid to minimize radiation exposure to workers during MPC closure operations. The MPC lid has a stepped recess around the perimeter for accommodating the closure ring. The MPC closure ring is welded to the MPC lid on the inner diameter of the ring and to the MPC shell on the outer diameter.

Following fuel loading and MPC lid welding, the MPC lid-to-shell weld is examined by progressive liquid penetrant examinations (a multi-layer liquid penetrant examination), and a pressure test is performed. The MPC lid-to shell weld is not helium leakage tested since the weld meets the guidance of NRC Interim Staff Guidance (ISG)-15 [7.1.2] and criteria of ISG-18 [7.0.2], therefore leakage from the MPC lid-to-shell weld is not considered credible. Table 7.1.2 provides the matrix of ISG-18 criteria and how the Holtec MPC design and associated inspection, testing, and QA requirements meet each one.

After the MPC lid weld is ensured to be acceptable the vent and drain port cover plates are *welded in place, examined by the liquid penetrant method and a helium leakage test of each of the vent and drain port cover plate welds is performed. These welds are tested in accordance to the leakage test methods and procedures of ANSI N 14.5 [7.0.3] to the "leaktight" criterion of the standard. Finally, the MPC closure ring which also covers the vent and drain cover plates is installed, welded, and inspected by the liquid penetrant method. Chapters 9, 10, and 13 provide procedural guidance, acceptance criteria, and operating controls, respectively, for performance and acceptance of non-destructive examination of all welds made in the field.*

After moisture removal and prior to sealing the MPC vent and drain ports, the MPC cavity is backfilled with helium. The helium backfill provides an inert, non-reactive atmosphere within the MPC cavity that precludes oxidation and hydride attack on the SNF cladding. Use of a helium atmosphere within the MPC contributes to the long-term integrity of the fuel cladding, reducing the potential for release of fission gas or other radioactive products to the MPC cavity. Helium also aids in heat transfer within the MPC and helps reduce the fuel cladding temperatures. The inert atmosphere in the MPC, in conjunction with the thermal design features of the MPC and storage cask, assures that the fuel assemblies are sufficiently protected against degradation, which might otherwise lead to gross cladding ruptures during long-term storage.

The confinement boundary welds completed at the fabrication facility (i.e., the MPC longitudinal and circumferential shell welds and the MPC shell to baseplate weld) are referred to as the shop welds. After visual and liquid penetrant examinations, the shop welds are volumetrically inspected by radiography. The MPC shop welds are multiple-pass (6 to 8 passes) austenitic stainless steel welds. Helium leakage testing of the shop welds is performed as described in Table 9.1.1.

7.1.2 Confinement Penetrations

Two penetrations (the MPC vent and drain ports) are provided in the MPC lid for MPC draining, moisture removal and backfilling during MPC loading operations, and also for MPC re-flooding during unloading operations. No other confinement penetrations exist in the MPC.

The MPC vent and drain ports are sealed by cover plates that are integrally welded to the MPC lid. No credit is taken for the sealing action provided by the vent and drain port cap joints. The MPC closure ring covers the vent and drain port cover plate welds and the MPC lid-to-shell weld, provide the redundant closure of these penetrations. The redundant closure of the MPC satisfies the requirements of 10CFR72.236(e) [7.0.1].

7.1.3 Seals and Welds

Section 7.1.1 describes the design of the confinement boundary welds. The welds forming the confinement boundary is summarized in Table 7.1.1.

The use of multi-pass welds with surface liquid penetrant inspection of root, intermediate, and final passes renders the potential of a leak path through the weld between the MPC lid and the shell to be non-credible. The vent and drain port cover plate welds are helium leak tested in the field, providing added assurance of weld integrity. Additionally after fuel loading, a Code *pressure test is performed on the MPC lid-to-shell weld to confirm the structural integrity of the weld.*

The ductile stainless steel material used for the MPC confinement boundary is not susceptible to delamination or other failure modes such as hydrogen-induced weld degradation. The closure weld redundancy assures that failure of any single MPC confinement boundary closure weld will not result in release of radioactive material to the environment. Section 10.1 *provides a summary of the closure weld examinations and tests.*

Although the MPC is credited with providing the confinement boundary under storage conditions, the HI-STAR 100 overpack provides an additional barrier against the release of radiological matter to the environment [1.2.6]. Concentric metallic seals in the overpack closure plate prevent the leakage of the helium gas from the annulus and provide an additional boundary to the release of radioactive materials. The seals on the overpack vent and drain port plugs are leak tested along with the overpack closure plate inner seal. Cover plates with metallic seals are installed over the overpack vent and drain ports to provide redundant closure of the overpack penetrations. A port plug with a metallic seal is installed in the overpack closure plate test port to provide fully redundant closure of all potential leakage paths in the overpack penetrations.

The HI-STAR 100 is a dual purpose cask, used for both storage and transport. Leakage Rate Acceptance Criteria for the overpack during transport is provided in Chapter 4 of Reference [1.2.6].

7.1.4 Closure

The MPC is an integrally welded pressure vessel without any unique or special closure devices. All closure welds are examined using the liquid penetrant technique to ensure their integrity. Additionally, the vent and drain port cover plate welds are each helium leakage tested to be "leaktight" in accordance with the leakage test methods and procedures of ANSI N14.5-1997 [7.0.3]. Since the MPC uses an entirely welded redundant closure system with no credible leakage, no direct monitoring of the closure is required.

Table 7.1.1 (INTENTIONALLY DELETED)

Table 7.1.2

MPC CONFINEMENT BOUNDARY WELDS

Confinement Boundary Welds		
MPC Weld Location	**Weld Type**†	**ASME Code Category (Section III, Subsection NB)**
Shell longitudinal seam	Full Penetration Groove (shop weld)	A
Shell circumferential seam	Full Penetration Groove (shop weld)	B
Baseplate to shell	Full Penetration Groove (shop weld)	C
MPC lid to shell	Partial Penetration Groove (field weld)	C
MPC closure ring to shell	Fillet (field weld)	††
Vent and drain port cover plates to MPC lid	Partial Penetration Groove (field weld)	D
MPC closure ring to closure ring radial	Partial Penetration Groove (field weld)	††
MPC closure ring to MPC lid	Partial Penetration Groove (field weld)	C

† The tests and inspections for the confinement boundary welds are listed in Section 9.1.1.
†† This joint is covered by NB-5271 (liquid penetrant examination).

Table 7.1.3 (INTENTIONALLY DELETED)

Table 7.1.4

COMPARISON OF HOLTEC MPC DESIGN WITH ISG-18 GUIDANCE

DESIGN/QUALIFICATION GUIDANCE	***HOLTEC MPC DESIGN***
The canister is constructed from austenitic stainless steel.	*The MPC enclosure vessel is constructed entirely from austenitic stainless steel (Alloy X). Alloy X is defined as Type 304, 304LN, 316, or 316LN material.*
The canister closure welds meet the guidance of ISG-15 (or approved alternative), Section X.5.2.3.	*The MPC lid-to-shell closure weld meets ISG-15, Section X.5.2.3 for austenitic stainless steels. UT examination is permitted and NB-5332 acceptance criteria are required. An optional multi-layer PT examination is also permitted. The multi-layer PT is performed at each approximately 3/8" of weld depth, which corresponds to the critical flaw size.*
The canister maintains its confinement integrity during normal conditions, anticipated occurrences, and credible accidents and natural phenomena as required in 10CFR72.	*The MPC is shown by analysis to maintain confinement integrity for all normal, off-normal, and accident conditions, including natural phenomena. The MPC is designed to ensure that the Confinement Boundary will not leak during any credible accident event and under the non-mechanistic tip-over scenario.*
Records documenting the fabrication and closure welding of canisters shall comply with the provisions 10CFR72.174 and ISG-15. Record storage shall comply with ANSI N45.2.9.	*Records documenting the fabrication and closure welding of MPCs meet the requirements of ISG-15 via controls required by the FSAR and HI-STAR 100 CoC. Compliance with 10CFR72.174 and ANSI N.45.2.9 is achieved via Holtec QA program and implementing procedures.*
Activities related to inspection, evaluation, documentation of fabrication, and closure welding of canisters shall be performed in accordance with an NRC-approved quality assurance program.	*The NRC has approved Holtec's Quality Assurance program under 10CFR71. That same QA program has been adopted for activities governed by 10CFR72 as permitted by 10 CFR 72.140(d)*

7.2 <u>REQUIREMENTS FOR NORMAL AND OFF-NORMAL CONDITIONS OF STORAGE</u>

Once sealed and transferred into the HI-STAR 100 *overpack there is no mechanism under normal and off-normal conditions of storage for the confinement boundary to be breached. Chapter 3 shows that all confinement boundary components are maintained within their Code-allowable stress limits during normal and off-normal storage conditions. Chapter 4 shows that the peak confinement boundary component temperatures and pressures are within the design basis limits for all normal and off-normal conditions of storage. Since the MPC confinement vessel remains intact, the design temperatures and pressure are not exceeded, and leakage from the MPC confinement boundary as discussed in Section 7.1 is not credible, there can be no release of radioactive material during normal and off-normal conditions of storage.*

The MPC is dried and helium backfilled prior to sealing and no significant moisture or other gases remain inside the MPC. Therefore, a credible mechanism for any radiolytic decomposition that could cause an increase in the MPC internal pressure is absent. The potential for the explosive level of gases due to radiological decomposition in the MPC is eliminated by excluding foreign materials in the MPC.

7.3 CONFINEMENT REQUIREMENTS FOR HYPOTHETICAL ACCIDENT CONDITIONS

The analysis in Chapter 3 and results discussed in Chapter 12 demonstrates that the MPC remains intact during and after all postulated accident conditions; therefore there can be no release of radioactive material causing any additional dose contribution to the site boundary during these events.

Table 7.3.1 – Table 7.3.6 (INTENTIONALLY DELETED)

7.4 REFERENCES

[7.0.1] 10CFR72, Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor Related Greater than Class C Waste," USNRC, Washington, DC.

[7.0.2] Interim Staff Guidance-18, "The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation," USNRC, Washington, DC, May 2003.

[7.0.3] ANSI N14.5-1997, "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment," American National Standards Institute, Washington, DC, 1997.

[7.0.4] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", USNRC, Washington, DC, January, 1997.

[7.0.5] Holtec Proprietary Report HI-2114830, Latest Revision, "Final Safety Analysis Report on The HI-STORM FW MPC Storage System" NRC Docket Number: 72-1032.

[7.1.1] ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Class 1 Components, American Society of Mechanical Engineers, New York, NY, 2007 Edition.

[7.1.2] Interim Staff Guidance-15, "Materials Evaluation", USNRC, Washington, DC, January 2001.

[7.1.3] Holtec Proprietary Report HI-2022850, Revision 0, "Summary Report on MPC Leak Tightness Test", April 2002.

CHAPTER 8: OPERATING PROCEDURES

Much of the new information in this chapter is directly extracted from previously NRC approved Holtec dockets; this information is shown in *italics*. In Chapter 8, this information was extracted from the HI-STORM 100 FSAR under HI-STORM 100 Storage Docket, 72-1014 [8.0.5] and the HI-STAR 100 SAR under the transportation Docket 71-9261 [8.0.6]. All changes in this revision are marked with revision bars[1].

8.0 INTRODUCTION:

This chapter outlines the loading, unloading, and recovery procedures for the HI-STAR 100 System for short term operations. *Short term operations* are defined as those activities that are required to change the storage environment from wet to dry, or vice versa. Short term operations also include preparing the cask to transport a loaded cask from an ISFSI or receiving a loaded cask at a storage site. This chapter outlines the short term operations applicable to the HI-STAR 100 cask with special emphasis on ALARA, personnel safety and Technical Specification compliance. The procedures outlined in this chapter are prescriptive to the extent that they provide the basis and general guidance to the user in preparing detailed written site-specific loading, handling, storage and unloading procedures. The user of HI-STAR 100 will utilize the procedures provided in this chapter, the Technical Specifications, the conditions of the Certificate of Compliance, equipment-specific operating instructions, and plant working procedures and apply them to develop the site-specific written loading, handling, unloading and storage procedures. Because the architectural features, thermo-physical and structural characteristics of the host sites vary widely, it is not possible to describe the short term operations for every scenario. The material in this chapter is therefore intended to spell out the permissible parameters of operations that are sanctioned by the safety analyses and evaluations that have been performed in this FSAR or in another NRC approved SAR or FSAR. Specific requirements and prohibitions that address worker as well as public health and safety to insure safe and ALARA use of the HI-STAR 100 cask are specifically addressed in this chapter. Likewise, measures and methods to respond to an abnormal or accident event are outlined in this chapter to enable a host site to evaluate its Emergency Preparedness program and make the necessary enhancements to its procedures to insure compliance with the prescriptive guidelines in this chapter.

The procedures contained herein describe acceptable methods for performing HI-STAR 100 loading and unloading operations. The generic procedures provided herein for the loading, storage, handling, and unloading of spent fuel in the HI-STAR 100 System are focused on minimizing the likelihood of an adverse occurrence that may impact crew health and safety. While the design of the HI-STAR 100 System and its ancillaries, the operating procedures summarized in this chapter, and the Technical Specifications serve to minimize risks and mitigate consequences of potential events, the activities involved in loading of spent fuel in a canister system, if not carefully performed, may nevertheless result in personnel hazards and radiological impact. In addition to well-articulated site specific procedures for normal loading

[1] Wherever multiple units are shown, the US units are the governing value, and the SI units are for information only.

and unloading operations, the plant must be in the possession of QA validated procedures to respond to abnormal and accident conditions. As a minimum, the procedures shall address establishing emergency action levels, implementation of emergency action program, establishment of personnel exclusion zones, monitoring of radiological conditions, actions to mitigate or prevent the release of radioactive materials, and recovery planning and execution.

The site specific procedures prepared by the user may alter the generic procedural steps provided in this chapter to allow the operations to be performed in parallel or out of sequence or may add or delete steps as long as the general intent of the information provided in this chapter is met. For example, dry loading and unloading of the system in a hot cell or other remote handling facility is permitted provided the implementing procedures will depart substantially from the standard wet transfer operation discussed in this chapter. Such procedures are acceptable for use if they comply with the restrictions and criteria set down in this chapter.

Holtec International maintains an active lessons learned database that should be consulted by the user in developing the operating procedures. This database contains a listing all human performance errors of omission and commission, equipment malfunctions and abnormal occurrences that should be used by the procedure developer to avoid repetition of past errors. All members of the Holtec User's Group (HUG) have proprietary access to the lessons learned database. Table 8.0.1 contains a catalog of those credible errors which may or may not have occurred in the past loading campaigns but are sufficiently important to warrant a careful review by the procedure developer.

The information provided in this chapter is intended to meet the guidelines of NUREG-1536 [8.0.1].

Table 8.0.1
CREDIBLE MISHAPS & HUMAN PERFORMANCE ERRORS IN A DRY STORAGE CAMPAIGN

	Risk	Comments
1	Safety of all applicable site specific hazards to short term operations not evaluated pursuant to 10CFR72.212	All credible natural phenomena germane to the host site, such as flood, tornado, hurricane, fire, earthquakes & the like, must be considered in the safety evaluation for deploying the HI-STAR casks both during short term operations and during long term storage. This FSAR considers the Design Basis Loads which may not bound the site specific loading in every case.
2	Loading fuel with incorrect Fuel Control Parameters (FCPs)	The enrichment, burn-up and cooling time restrictions on the fuel (Fuel Control Parameters) to be loaded in each specific storage location must be carefully evaluated against the restrictions delineated in the Technical Specifications for the system.
3	*Damaged fuel* loaded bare (without DFCs)	Each fuel assembly to be loaded in the MPC must be characterized to insure that those classified as *damaged* are installed in a Damaged Fuel Canister (DFC). The DFCs prevent the particulate matter released by the damaged pellets (if any) from migrating outside the storage cell.
4	Violation of the limits on the Fuel Control (FCPs) Parameters for the DFCs	The limits on the fuel control parameters (e.g., fuel type, enrichment, burn-up & cooling time) and the eligible storage locations applicable to the DFC must be carefully aligned with the Technical Specification.
5	Handling accident in the Fuel Building	The safety evaluation of all credible accidents during the cask's handling within the part 50 facility is mandatory.
6	Lifted weight exceeds the capacity of the cask's trunnions	Users shall implement controls to ensure that the maximum lifted weights do not exceed the HI-STAR 100 trunnion design limits.
7	Boiling of water in the MPC	Users shall implement controls to monitor the time limit from the removal of the HI-STAR 100 from the spent fuel pool to the commencement of MPC draining to prevent boiling.
4	Excessive radiation exposure during fuel loading operations in the Fuel Building	Expected radiation levels during loading shall be computed and auxiliary shielding applied to ensure ALARA.
5	Generation of excessive radioactive waste	The HI-STAR 100 System uses design features that minimize the amount of radioactive waste generated. Such features include smooth surfaces for ease of decontamination efforts, prevention of avoidable contamination, and procedural guidance to reduce decontamination requirements. Measures shall be implemented to prevent excessive rad-waste generation.
6	Contamination of the MPC's external surface	A proven annulus over-pressure system along with a flexible annulus closure shall be employed to sequester the external surfaces of the MPC from pool's water. The MPC will be rid of any contaminant that may have thwarted the isolation system by a suitable means such as hydro-lasing.
7	Damage to the vacuum gage from excessive positive pressure	The vacuum gauges, used in the Vacuum Drying System, should never be allowed to be exposed to pressure above their recommended range.

8	Restrictions on loading of High Burnup Fuel	The loading operations shall be configured to insure that the more restrictive temperature limits on High Burnup fuel set down in ISG-11 Rev 3 shall not be violated.
9	Dilution of pool water	Loading of fuel in PWR MPCs (viz., MPC-32 & MPC-24) requires lower threshold limit on the boron concentration set down in the Technical Specification. Inadvertent dilution of pool water is a credible operational error which calls for a strong preventive measure.
10	Ignition of combustible mixture during welding	Hydrogen generation inside the MPC, caused by the chemical reaction of the water with a reactive and porous material such as Boral or with the paint applied to carbon steel baskets have been previously identified as sources of hydrogen ignition sparked by the arc during the canister's welding. To prevent a hydrogen ignition event, Holtec MPCs don't employ any carbon steel exposed to pool water, or Boral. In addition, the underneath the lid is required to be purged of any gas generated within the MPC throughout the duration of the welding operation.
11	Ambient temperature too high in the Fuel building	The temperature of the fuel cladding in the loaded MPC is directly affected by the ambient temperature in the Fuel Building. Plants located in the southern latitudes must consider the effect of the elevated ambient temperature on the compliance of the cladding temperature with ISG-11 Rev 3 limits.
12	Fuel building ambient too cold-risk of Brittle fracture in lifting equipment	The ambient temperature in the Fuel Building in plants located in the northern latitudes during winter months may be sufficiently low to cause brittle fracture concerns in the thick steel sections of the lifting and handling devices. The risk of brittle fracture during handling of the cask in such cold conditions must be evaluated and determined to be precluded before such an operation is permitted to occur.
13	Violation of ISG-11 fuel cladding temperature limits and temperature excursions during vacuum drying	The state of near vacuum in the MPC caused during the Vacuum Drying operation is thermally the most severe condition for the used fuel. ISG-11 Rev 3 provides stringent temperature limits on the fuel cladding under drying conditions. Compliance with the ISG-11 Rev 3 temperature limits and temperature excursion limits is mandated by this FSAR. To eliminate the risk of non-compliance, the Forced Helium Dehydration (FHD) system may be used.
14	Contamination of cask's external surface from exposure to pool water	Depending on the cleanliness condition of the pool, it may be necessary to utilize a prophylactic film or other means to protect (most of) the cask's surface from coming in direct contact with the pool water.
15	Fire during cask's on-site transportation	A complete assay of all sources of combustion near the haul path and removing those with a large energy content is a mandatory pre-loading activity. In addition, the quantity of diesel carried in the transporter's tank shall be limited to the value specified in this FSAR. As defense-in-depth, a fire event wherein all flammable materials participate (including the transmission fluid) in the fire shall be performed to insure that the limits to insure fuel integrity and radiation blockage effectiveness of the cask set forth in this FSAR, will not be violated.
16	Mis-operation of the Mating Device	The Mating Device's drawer shall be verified to be operable- under-load prior to the start of the loading campaign.
17	Mis- operation of the transporter	All potential failure modes for the type of transporter used in the loading campaign, based on prior loading experience, shall be evaluated and remedial measures implemented to avoid an unplanned interruption

		during the loading campaign.
18	Seismic stability of the cask in the fuel building during short term operations	The kinematic stability of the loaded cask during its various staged evolutions shall be evaluated using Holtec's established analysis procedures. Such evolutions may include the cask arrayed on the pool slab (cask pit), in the decontamination and welding station and in the truck bay. In general, a nuclear plant's work rules don't require evaluation of seismic stability of a load while is being carried by the cask crane inside the Fuel Building on PRA grounds. The standard practice followed by the plant in this matter will govern cask operations inside the part 50 structure.
19	Seismic stability of the loaded transporter	The loaded transporter shall be shown by appropriate analysis, to have at least 10% margin against tip-over under the site's Design Basis Earthquake.
20	Handling accident on the haul path or at the ISFSI	In addition to adopting pro-active measures to prevent a cask handling accident, the occurrence of all credible handling accidents shall be considered. The analysis shall show, at minimum, that the MPC shall not be ejected outside its biological shield (cask).

8.1 FUEL LOADING OF THE HI-STAR 100 SYSTEM

8.1.1 Information to Support Loading Operations

Tables 8.1.1 and 8.1.2 provide the handling weights for each of the HI-STAR 100 System major components and the loads to be lifted during the operation of the HI-STAR 100 System. Table 8.1.3 provides the HI-STAR 100 System bolt torque and sequencing requirements. Table 8.1.4 provides an operational description of the HI-STAR 100 System ancillary equipment and its safety designation. Fuel assembly selection and verification shall be performed by the licensee in accordance with written, approved procedures which ensure that only SNF assemblies authorized in Appendix B to the Certificate of Compliance are loaded into the HI-STAR 100 System.

In addition to the requirements set forth in the CoC, users will be required to develop or modify existing programs and procedures to account for the operation of an ISFSI. Written procedures are required to be developed or modified to account for such things as nondestructive examination (NDE) of the MPC welds, handling and storage of SSCs identified as Important to Safety, 10CFR72.48 [8.0.2] programs, specialized instrument calibration, special nuclear material accountability at the ISFSI, security modifications, fuel handling procedures, training and emergency response, equipment and process qualifications. Table 8.1.5 summarizes the instrumentation used to load and unload the HI-STAR 100 System. Tables 8.1.6 and 8.1.7 provide sample receipt inspection checklists for the HI-STAR 100 overpack and the MPC, respectively. Users shall develop site-specific receipt inspection checklists, as required. Fuel handling, including the handling of fuel assemblies in the Damaged Fuel Container (DFC) shall be performed in accordance with written site-specific procedures. Damaged fuel and fuel debris, as defined in the Technical Specifications appended to CoC 1008 shall be loaded in DFCs.

8.1.2 Overview of Loading Operations

The HI-STAR 100 System is used to load, unload, transfer and store spent fuel. Specific steps are performed to prepare the HI-STAR 100 System for fuel loading, to load the fuel, to prepare the system for storage and to place it in storage at an ISFSI. The HI-STAR 100 overpack may be transferred between the ISFSI and the fuel loading facility using a specially designed transporter, heavy haul transfer trailer, or any other load handling equipment designed for such applications as long as the lifting requirements described in Chapter 1 and the Technical Specifications are met. Users shall develop detailed written procedures to control on-site transport operations. Section 8.1.2 provides the general procedures for handling of the HI-STAR 100 overpack and MPC. Figure 8.1.1 shows a flow diagram of the HI-STAR 100 System loading operations. Figure 8.1.2 illustrates some of the major HI-STAR 100 System loading operations.

Note:
The procedures describe plant facilities, functions, and processes in general terms. Each site is different with regard to layout, organization and nomenclature. Users shall interpret the nomenclature used herein to suit their particular site, organization, and methods of operation.

Refer to the boxes of Figure 8.1.2 for the following description. At the start of loading operations, an empty MPC is upended (Box 1). The empty MPC is raised and inserted into the

HI-STAR 100 overpack (Box 2). The annulus is filled with plant demineralized water and the MPC is filled with either spent fuel pool water or plant demineralized water (Box 3). An inflatable seal is installed in the annulus between the MPC and the HI-STAR 100 overpack to prevent spent fuel pool water from contaminating the exterior surface of the MPC. The HI-STAR 100 overpack and the MPC are then raised and lowered into the spent fuel pool for fuel loading using the lift yoke (Box 4). Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed (Box 5).

While still underwater, a thick, shielded lid (the MPC lid) is installed using either slings attached to the lift yoke or the Lid Retention System (Box 6). The lift yoke remotely engages to the HI-STAR 100 overpack lifting trunnions to lift the HI-STAR 100 overpack and loaded MPC close to the spent fuel pool surface (Box 7). When radiation dose rate measurements confirm that it is safe to remove the HI-STAR 100 overpack from the spent fuel pool, the cask is removed from the spent fuel pool. If the Lid Retention System is being used, the HI-STAR 100 overpack closure plate bolts are installed to secure the MPC lid for the transfer to the cask preparation area. The lift yoke and HI-STAR 100 overpack are sprayed with demineralized water to help remove contamination as they are removed from the spent fuel pool.

The HI-STAR 100 overpack is placed in the designated preparation area and the lift yoke and Lid Retention System retention disk are removed. The next phase of decontamination is then performed. The top surfaces of the MPC lid and the upper flange of the HI-STAR 100 overpack are decontaminated. The Temporary Shield Ring (if utilized) is installed and filled with water. The inflatable annulus seal is removed, and the annulus shield is installed. The Temporary Shield Ring provides additional personnel shielding around the top of the HI-STAR 100 overpack during MPC closure operations. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped into the annulus. Dose rates are measured at the MPC lid and around the mid-height circumference of the HI-STAR 100 overpack to ensure that the dose rates are within expected values.

The MPC water level is lowered slightly, the MPC is vented, and the MPC lid is seal welded using the Automated Welding System (Box 8). Visual examinations are performed on the tack welds. Liquid penetrant examinations are performed on the root and final passes. An ultrasonic or multi-layer PT examination is performed on the MPC Lid-to-Shell weld to ensure that the weld is satisfactory. As an alternative to volumetric examination of the MPC lid-to-shell weld, a multi-layer PT is performed including one intermediate examination after approximately every three-eighth inch of weld depth. *The MPC welds are then pressure tested followed by an additional liquid penetrant examination performed on the MPC Lid-to-Shell weld to verify structural integrity. To calculate the helium backfill requirements for the MPC (if backfill is based upon helium mass or volume measurements), the free volume inside the MPC must first be determined. This free volume may be determined by measuring the volume of water displaced or any other suitable means.*

Caution:
Inert gas must be used any time the fuel is not covered with water to prevent oxidation of the fuel cladding. The fuel cladding is not to be exposed to air at any time during loading operations.

The water level is raised to the top of the MPC again and then the MPC water is displaced from the MPC by blowdown of the water using pressurized helium or nitrogen gas introduced into the vent port of the MPC thus displacing the water through the drain line. The Vacuum Drying System (VDS) is connected to the MPC and is used to remove all residual liquid water from the MPC in a stepped evacuation process (Box 9). A stepped evacuation process is used to preclude the formation of ice in the MPC and Vacuum Drying System lines. The internal pressure is reduced to below 3 torr (400 Pa) and held for 30 minutes to ensure that all liquid water is removed.

Following the dryness test, the VDS is disconnected, the Helium Backfill System (HBS) is connected, and the MPC is backfilled with a predetermined pressure of helium gas as specified in the Technical Specification. The helium backfill ensures adequate heat transfer during storage, provides an inert atmosphere for long-term fuel integrity. Cover plates are installed and seal welded over the MPC vent and drain ports and liquid penetrant examinations are performed on the root (for multi-pass welds) and final passes (Box 10). The cover plates are then leak tested.

Alternatively, the Forced Helium Dehydration System (FHD), licensed by the NRC in several Holtec dockets may be used to remove residual moisture from the MPC. Gas is circulated through the MPC to evaporate and remove moisture. The residual moisture is condensed until no additional moisture remains in the MPC. The temperature of the gas exiting the system demoisturizer is maintained below 21ºF (-6.11°C) for a minimum of 30 minutes to ensure that all liquid water is removed. The MPC is then backfilled to the required helium backfill pressure. The FHD has been approved by the NRC with MPC-24, -68, and -32 under SER Revision 8, dated October 12, 2010.

The MPC closure ring is then placed on the MPC and dose rates are measured at the MPC lid to ensure that the dose rates are within expected values. The closure ring is aligned, tacked in place and seal welded providing redundant closure of the MPC confinement boundary closure welds. Tack welds are visually examined, and the root (for multi-pass welds) and final welds are inspected using the liquid penetrant examination technique to ensure weld integrity.

The annulus shield is removed and the remaining water in the annulus is drained. The MPC lid and accessible areas at the top of the MPC shell are smeared for removable contamination and the HI-STAR 100 overpack dose rates are measured to establish compliance with the Technical Specification limits. The HI-STAR 100 overpack closure plate is installed (Box 11) and the bolts are torqued. The HI-STAR 100 overpack annulus is vacuum dried and backfilled with helium gas as specified in the Technical Specification. The HI-STAR 100 overpack mechanical seals are helium leakage tested to assure they will provide long-term retention of the annulus helium. The HI-STAR 100 overpack cover plates are installed. The Temporary Shield Ring is drained and removed. Dose rates are taken on the overpack to ensure that they are below the Technical Specification limits.

The HI-STAR 100 overpack is moved to the ISFSI pad (Box 12). The HI-STAR 100 overpack may be moved using a number of methods as long as the lifting requirements of set forth in Chapter 1 and the Technical Specifications are met.

8.1.3 <u>HI-STAR 100 System Receiving and Handling Operations:</u>

Note:
The HI-STAR 100 overpack may be received and handled in several different configurations and may be transported on-site in a horizontal or vertical orientation. This section provides general guidance for the HI-STAR 100 overpack and MPC rigging and handling. Site-specific procedures shall specify the required operational sequences based on the cask handling configuration and limitations at the sites. Steps 1 through 4 describe the handling operations using a lift yoke. Specialty rigging may be substituted if the lift complies with NUREG-0612 [8.0.4].

1. Vertical Handling of the HI-STAR 100 overpack:

   a. Verify that the lift yoke load test certifications are current.

   b. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage. *Replace or repair damaged components as necessary.*

   c. Engage the lift yoke to the lifting trunnions. See Figure 8.1.3.

   d. Apply lifting tension to the lift yoke and verify proper engagement of the lift yoke.

Note:
Refer to the site's heavy load handling procedures for lift height, load path, floor loading and other applicable load handling requirements. Ensure that the requirements on lifting devices in Chapter 1 and the Technical Specifications are met.

   e. Raise the HI-STAR 100 overpack and position it accordingly.

2. Upending of the HI-STAR 100 overpack in the transport frame or rotation cradle:

Warning:
Personnel shall remain clear of the unshielded bottom of the loaded overpack to the maximum extent possible. Keeping the bottom region of the cask overlaid with temporary shielding during the cask’s up-ending is a necessary ALARA requirement.

   a. If installed, remove the overpack bottom cover. Rigging points are provided. See Figure 8.1.4.

   b. Position the HI-STAR 100 overpack under the lifting device. Refer to Step 1, above.

   c. Verify that the lift yoke load test certifications are current.

   d. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage.

e. Engage the lift yoke to the lifting trunnions. (The use of a ratchet strap or similar device to restrain the lift yoke arms is recommended during HI-STAR upending operation). See Figure 8.1.3.

f. Apply lifting tension to the lift yoke and verify proper engagement of the lift yoke.

g. Slowly rotate the HI-STAR 100 overpack to the vertical position keeping all rigging as close to vertical as practicable. See Figure 8.1.4.

h. Lift the pocket trunnions, if used, clear of the transport frame rotation trunnions. If the cask tilting plate is installed on the HI-STAR 100, lift the cask tilting plate trunnions clear of the rotation cradle.

i. Position the HI-STAR 100 overpack per site direction.

3. Downending of the HI-STAR 100 overpack in the transport frame or rotation cradle:

a. Position the transport frame or rotation cradle under the lifting device.

b. Verify that the lift yoke load test certifications are current.

c. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage.

d. Engage the lift yoke to the lifting trunnions. (The use of a ratchet strap or similar device to restrain the lift yoke arms is recommended during HI-STAR downending operation). See Figure 8.1.3.

e. Apply lifting tension to the lift yoke and verify proper lift yoke engagement.

f. Position the pocket trunnions, if used, to receive the transport frame rotation trunnions. See Figure 8.1.4. If the cask tilting plate is installed on the HI-STAR 100, position the rotation frame trunnions to receive the rotation cradle.

g. Slowly rotate the HI-STAR 100 overpack to the horizontal position keeping all rigging as close to vertical as practicable.

h. Disengage the lift yoke.

Warning:
Personnel shall remain clear of the unshielded bottom of the loaded overpack to the maximum extent possible. Keeping the bottom region of the cask overlaid with temporary shielding during the cask's downending is a necessary ALARA requirement.

i. If necessary for radiation shielding, install the overpack bottom cover. Rigging points are provided. See Figure 8.1.4.

4. Horizontal Handling of the HI-STAR 100 overpack in the transport frame or rotation cradle:

    a. Secure the transport frame or rotation cradle for HI-STAR 100 downending.

    b. Downend the HI-STAR 100 overpack on the transport frame or rotation cradle per Step 3, if necessary.

    c. Inspect the transport frame or rotation cradle lift rigging in accordance with site approved rigging procedures.

    d. Position the transport frame or rotation cradle accordingly.

5. Empty MPC Installation in the HI-STAR 100 overpack:

Note:
To avoid side loading the MPC lift lugs, the MPC should be upended in the MPC Upending Frame (or equivalent).

    a. If necessary, remove any MPC shipping covers and rinse off any road dirt with water. Be sure to remove any foreign objects from the MPC internals.

    b. If necessary, upend the MPC as follows:

        1. Visually inspect the MPC Upending Frame for gouges, cracks, deformation or other indications of damage. *Repair or replace damaged components as necessary.*

        2. Install the MPC on the Upending Frame. Make sure that the banding straps are secure around the MPC shell. See Figure 8.1.5.

        3. Inspect the Upending Frame slings in accordance with the site's lifting equipment inspection procedures. Rig the slings around the bar in a choker configuration to the outside of the cleats. See Figure 8.1.5.

        4. Attach the MPC upper end slings of the Upending Frame to the main overhead lifting device. Attach the bottom-end slings to a secondary lifting device (or a chain fall attached to the primary lifting device).

        5. Raise the MPC in the Upending Frame.

        6. Slowly lift the upper end of the Upending Frame while lowering the bottom end of the Upending Frame.

        7. When the MPC approaches the vertical orientation, release the tension on the lower slings.

        8. Place the MPC in a vertical orientation on a level surface.

9. Disconnect the MPC straps and disconnect the rigging.

c. Install the MPC in the HI-STAR 100 overpack as follows:

1. Install the four point lift sling to the lift lugs inside the MPC.

Caution:
The seal seating surface of the cask shall be protected from damage during MPC installation.

2. Raise and place the MPC inside the HI-STAR 100 overpack.

Note:
An alignment punch mark is provided on the HI-STAR 100 overpack and the top edge of the MPC. Similar marks are provided on the MPC lid and closure ring.

3. Rotate the MPC so the alignment marks agree and seat the MPC inside the HI-STAR 100 overpack. Disconnect the MPC rigging or the MPC lift rig.

8.1.4 <u>HI-STAR 100 Overpack and MPC Receipt Inspection and Loading Preparation</u>

ALARA Note:
A bottom protective cover may be attached to the HI-STAR 100 overpack bottom or placed in the designated preparation area and spent fuel pool. This will help prevent embedding contaminated particles in the HI-STAR 100 overpack bottom surface and ease the decontamination effort.

1. Receipt Inspection

a. Place the HI-STAR 100 overpack in the cask receiving area. Perform appropriate contamination and security surveillances, as required.

1. If necessary, remove the HI-STAR 100 overpack closure plate by removing the closure plate bolts. See Figure 8.1.8 for rigging example.

2. Place the closure plate on cribbing that protects the seal seating surfaces and allows access for seal replacement.

3. Install the seal surface protector on the HI-STAR 100 overpack seal seating surface.

b. Rinse off any road dirt with water. Inspect all cavity locations for foreign objects. Remove any foreign objects.

c. Disconnect the rigging.

d. Store the closure plate and bolts in a site-approved location.

e. At the site's discretion, perform an MPC receipt inspection and cleanliness inspection in accordance with a site-specific inspection checklist.

f. Install the MPC inside the HI-STAR 100 overpack and place the HI-STAR 100 overpack in the designated preparation area. See Section 8.1.2.

2. Install the upper fuel spacers in the MPC lid as follows:

> **Note:**
> Fuel spacers are fuel-type specific. Not all fuel types require fuel spacers. Upper fuel spacers are threaded into the underside of the MPC lid. Fuel spacers may be loaded any time prior to insertion of the fuel assemblies in the MPC.

a. Position the MPC lid on supports to allow access to the underside of the MPC lid.

b. Thread the fuel spacers into the holes provided on the underside of the MPC lid. See Table 8.1.3 for torque requirements.

c. Install threaded inserts in the MPC lid where and when spacers will not be installed, if necessary. See Table 8.1.3 for torque requirements.

3. Perform an MPC lid and closure ring fit test:

a. Visually inspect the MPC lid rigging (See Figure 8.1.8).

b. Raise the MPC lid such that the drain line can be installed. Install the drain line to the underside of the MPC lid.

> **Note:**
> The MPC Shell is relatively flexible compared to the MPC Lid and may create areas of local contact that impede Lid insertion in the Shell. Grinding of the MPC Lid below the minimum diameter on the drawing is permitted to alleviate interference with the MPC Shell in areas of localized contact. If the amount of material removed from the surface exceeds 1/8" (3.175 mm), the surface shall be examined by a liquid penetrant method (NB-2546). The weld prep for the Lid-to-Shell weld shall be maintained after grinding.

c. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC drain location. Verify that the MPC lid fit and weld prep are in accordance with the approved design drawings.

d. Install, *align and fit up* the closure ring.

e. Verify that closure ring fit and weld prep are in accordance with the approved fabrication drawings or the approved design drawings.

f. Remove the closure ring, *vent and drain port cover plates*, and the MPC lid. Disconnect the drain line. Store these components in an approved plant storage location.

4. *At the user's discretion, perform an MPC vent and drain port cover plate fit test and verify that the weld prep is in accordance with the approved fabrication drawings.*

Note:
Fuel spacers are fuel-type specific. Not all fuel types require fuel spacers. Lower fuel spacers are set in the MPC cells manually. No restraining devices are used. Fuel spacers may be loaded any time prior to insertion of the fuel assemblies in the MPC.

5. Install lower fuel spacers in the MPC (if required for the fuel type).

6. Fill the MPC and annulus as follows:

Caution:
Do not use any sharp tools or instruments to install the inflatable seal. Some air in the inflatable seal helps in the installation.

   a. Remove the HI-STAR 100 overpack drain port cover and port plug and install the drain connector. Store the drain port cover plate and port plug in an approved storage location.

   b. Fill the annulus with plant demineralized water to just below the inflatable seal seating surface.

   c. Manually insert the inflatable annulus seal around the MPC. Ensure that the seal is uniformly positioned in the annulus area

   d. Inflate the seal.

   e. Visually inspect the seal to ensure that it is properly seated in the annulus. Deflate, adjust and inflate the seal as necessary. Replace the seal as necessary.

ALARA Note:
Waterproof tape placed over empty bolt holes, and bolt plugs may reduce the time required for decontamination.

   f. At the user's discretion, install the HI-STAR 100 overpack closure plate bolt plugs and/or apply waterproof tape over any empty bolt holes.

ALARA Note:
Keeping the water level below the top of the MPC prevents splashing during handling.

   g. Fill the MPC with either demineralized water or spent fuel pool water to approximately 12 inches (30.5 cm) below the top of the MPC shell. *Refer to Tables 2.1.18 and 2.1.19 for required soluble boron concentration requirements.*

7. Place the HI-STAR 100 overpack in the spent fuel pool as follows:

ALARA Note:
The optional Annulus Overpressure System is used to provide further protection against MPC external shell contamination during in-pool operations.

a. If used, fill the Annulus Overpressure System lines and reservoir with demineralized water and close the reservoir valve. Attach the Annulus Overpressure System to the HI-STAR 100 overpack.

b. Verify spent fuel pool for boron concentration requirements in accordance with Tables 2.1.18 and 2.1.19.

c. Engage the lift yoke to the HI-STAR 100 overpack lifting trunnions and position the HI-STAR 100 overpack over the cask loading area with the basket aligned to the orientation of the spent fuel racks.

ALARA Note:
Pre-wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

d. Wet the surfaces of the HI-STAR 100 overpack and lift yoke with plant demineralized water while slowly lowering the HI-STAR 100 overpack into the spent fuel pool.

e. When the top of the HI-STAR 100 overpack reaches the elevation of the reservoir, open the Annulus Overpressure System reservoir valve. Maintain the reservoir water level at approximately 3/4 full the entire time the cask is in the spent fuel pool.

f. Place the HI-STAR 100 overpack on the floor of the cask loading area and disengage the lift yoke. Visually verify that the lift yoke is fully disengaged. Remove the lift yoke from the spent fuel pool while spraying the crane cables and yoke with plant demineralized water.

8.1.5 MPC Fuel Loading

Note:
When loading MPCs requiring soluble boron, the boron concentration of the water shall be checked in accordance with Tables 2.1.18 and 2.1.19 before and during operations with fuel and water in the MPC.

1. Perform a fuel assembly selection verification using plant fuel records to ensure that only fuel assemblies that meet all the conditions for loading as specified in Appendix B to Certificate of Compliance 1008 have been selected for loading into the MPC.

2. Load the pre-selected fuel assemblies into the MPC in accordance with the approved fuel loading pattern.

3. Perform a post-loading visual verification of the assembly identification to confirm that the serial numbers match the approved fuel loading pattern.

8.1.6 MPC Closure Operations

Note:
The user may elect to use the optional Lid Retention System to assist in the installation of the MPC lid and attachment of the lift yoke, and to provide the means to secure the MPC lid in the event of a drop or tip-over accident during loaded cask handling operations outside of the spent fuel pool. The effect of the additional hardware on the additional weight imposed on the cask, lift yoke, crane and floor prior to use shall be considered to ensure that crane capacity is not exceeded and heavy loads handling restrictions are observed. See Tables 8.1.1 and 8.1.2.

1. Visually inspect the MPC lid rigging or Lid Retention System in accordance with site-approved rigging procedures. Attach the MPC lid to the lift yoke so that MPC lid, drain line and trunnions will be in relative alignment. Raise the MPC lid and adjust the rigging so the MPC lid hangs level as necessary.

2. Install the drain line to the underside of the MPC lid.

3. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC drain location and the cask trunnions will also engage.

4. Slowly lower the MPC lid into the pool and insert the drain line into the drain access location and visually verify that the drain line is correctly oriented.

5. Lower the MPC lid while monitoring for any hang-up of the drain line. If the drain line becomes kinked or disfigured for any reason, remove the MPC lid and replace the drain line.

Note:
The upper surface of the MPC lid will seat approximately flush with the top edge of the MPC shell when properly installed. Once the MPC lid is installed, the HI-STAR/MPC removal from the spent fuel pool should proceed in a continuous manner to minimize the rise in MPC water temperature.

6. Seat the MPC lid in the MPC and visually verify that the lid is properly installed.

7. Engage the lift yoke to the HI-STAR 100 overpack lifting trunnions.

8. Apply a slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the lifting trunnions.

ALARA Note:
Activated debris may have settled on the top face of the HI-STAR 100 overpack and MPC during fuel loading. The cask top surface should be kept under water until a preliminary dose rate scan clears the cask for removal. *Users are responsible for any water dilution considerations.*

9. Raise the HI-STAR 100 overpack until the MPC lid is just below the surface of the spent fuel pool. Survey the area above the cask lid to check for hot particles. Raise and flush the upper surface of the HI-STAR 100 overpack and MPC with the plant demineralized water hoses as necessary to remove any activated particles from the HI-STAR 100 overpack or the MPC lid.

10. Visually verify that the MPC lid is properly seated. Lower the HI-STAR 100 overpack, reinstall the MPC lid, and repeat as necessary.

11. If the Lid Retention System is used, inspect the closure plate bolts for general condition. Replace worn or damaged bolts with new bolts.

12. Install the Lid Retention System bolts if the Lid Retention System is used.

13. If necessary for lifted weight conditions, pump a measured amount of water from the MPC. See and Tables 8.1.1 and 8.1.2.

14. Continue to raise the HI-STAR 100 overpack under the direction of the plant's radiological control personnel. Continue rinsing the surfaces with demineralized water. When the top of the HI-STAR 100 overpack reaches the approximate elevation as the reservoir, close the Annulus Overpressure System reservoir valve.

Caution:
Users are required to take necessary actions to prevent boiling of the water in the MPC. This may be accomplished by performing a site-specific analysis to identify a time limitation to ensure that water boiling will not occur in the MPC prior to the initiation of draining operations. Chapter 4 of the FSAR provides some sample time limits for the time to initiation of draining for various spent fuel pool water temperatures using design basis heat loads. These time limits may be adopted if the user chooses not to perform a site-specific analysis. If time limitations are imposed, users shall have appropriate procedures and equipment to take action if time limits are approached or exceeded. One course of action involves initiating an MPC water flush for a certain duration and flow rate. Any site-specific analysis shall identify the methods to respond should it become likely that the imposed time limit could be exceeded. *Refer to Tables 2.1.18 and 2.1.19 for boron concentration requirements whenever water is added to the loaded MPC.*

ALARA Note:
To reduce decontamination time, the surfaces of the HI-STAR 100 overpack and lift yoke should be kept wet until decontamination begins.

15. Remove the HI-STAR 100 overpack from the spent fuel pool while spraying the surfaces with plant demineralized water. Record the time.

ALARA Note:
Decontamination of the HI-STAR 100 overpack bottom should be performed using *remote cleaning methods, covering or other methods to minimize personnel exposure. .*

16. Decontaminate the HI-STAR 100 overpack bottom and perform a contamination survey of the HI-STAR 100 overpack bottom. Remove the bottom protective cover, if used.

17. If used, disconnect t the Annulus Overpressure System from the HI-STAR 100 overpack.

18. Set the HI-STAR 100 overpack in the designated cask preparation area.

19. Disconnect the lifting slings or Lid Retention System (if used) from the MPC lid and disengage the lift yoke. Decontaminate and store these items in an approved storage location.

Warning:
MPC lid dose rates are measured to ensure that dose rates are within expected values. Dose rates exceeding the expected values may indicate that fuel assemblies not meeting the specifications of the CoC have been loaded.

20. Measure the dose rates at the MPC lid and verify that the combined gamma and neutron dose rate is below the expected values.

21. Perform decontamination of the HI-STAR 100 overpack.

22. Prepare the MPC for MPC lid welding as follows:

ALARA Note:
If the Temporary Shield Ring is not used, some form of gamma shielding (e.g. lead bricks or blankets) should be placed in the areas above the HI-STAR neutron shield to eliminate the localized hot spot.

    a. Decontaminate the area around the HI-STAR 100 overpack top flange and install the Temporary Shield Ring, (if used).

    b. Fill the Temporary Shield Ring with water (if used).

    c. Carefully decontaminate the MPC lid top surface and the shell area above the inflatable annulus seal.

    d. Deflate and remove the annulus seal.

ALARA Note:
The water in the HI-STAR 100 overpack-to-MPC annulus provides personnel shielding. The level should be checked periodically and refilled accordingly.

23. Attach the drain line to the HI-STAR 100 overpack drain port connector and lower the annulus water level approximately 6 inches (15.2 cm).

ALARA Note:
The MPC exterior shell survey is performed to evaluate the performance of the inflatable annulus seal. Indications of contamination could require the MPC to be unloaded.

24. Survey the MPC lid top surfaces and the accessible areas of the top two inches (5 cm) of the MPC shell for surface contamination in accordance with the requirements of the Technical Specification.

ALARA Note:
The annulus shield is used to prevent objects from being dropped into the annulus and helps reduce dose rates directly above the annulus region. The annulus shield is hand installed and requires no tools.

25. Install the annulus shield. Prepare for MPC lid welding as follows:

Note:
The following steps use two identical Removable Valve Operating Assemblies (RVOAs) to engage the MPC vent and drain ports. The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage during drying, and to withstand the long-term effects of temperature and radiation. The RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operations. The RVOAs are purposely not installed until the cask is removed from the spent fuel pool to reduce the amount of decontamination.

Note:
The vent and drain ports are opened by pushing the RVOA operating rod down to engage the drive mechanism on the MPC port connection and turning the operating rod fully in the counter-clockwise direction. The operating rod will not turn once the port is fully open. Similarly, the vent and drain ports are closed by turning the operating rod fully in the clockwise direction. The ports are closed when the operating rod cannot be turned further.

a. Clean the vent and drain ports to remove any dirt. Install the RVOAs to the vent and drain ports leaving caps open.

ALARA Warning:
Personnel should remain clear of the drain lines any time water is being pumped or purged from the MPC. Assembly crud, suspended in the water, may create a radiation hazard to workers. Controlling the amount of water pumped from the MPC prior to welding keeps the fuel assembly cladding covered with water yet still allows room for thermal expansion.

b. Attach the water pump to the drain port and pump between 50 (189 L) and 120 gallons (454 L) of MPC water to the spent fuel pool or liquid radwaste system. The water level is lowered to keep moisture away from the weld region.

c. Disconnect the water pump.

26. Weld the MPC lid as follows:

Caution:

Grinding of MPC welds may create the potential for contamination. All grinding activities shall be performed under the direction of radiation protection personnel.

Oxidation of Boral panels contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during MPC lid welding operations. The space below the MPC lid shall be exhausted or purged with inert gas prior to, and during MPC lid welding operations to provide additional assurance that flammable gas concentrations will not develop in this space.

a. Attach a vacuum source to the vent port or inert the gas space under the MPC lid and begin monitoring for combustible gas concentrations.

b. If necessary center the lid in the MPC shell using a hand-operated chain fall.

Note:

The MPC is equipped with lid shims that serve to close the gap in the joint for MPC lid closure weld.

c. As necessary, install the MPC lid shims around the MPC lid to make the weld gap uniform.

ALARA Note:

The optional AWS Baseplate shield is used to further reduce the dose rates to the operators working around the top cask surfaces.

d. Install the Automated Welding System baseplate shield (if used). See Figure 8.1.8 for rigging.

e. Install the Automated Welding System Robot (if used). See Figure 8.1.8 for rigging.

f. Perform the MPC Lid-to-Shell weld and NDE with approved procedures. (See 9.1 and Table 2.2.15)

g. Disconnect the vacuum /purge source from the MPC and terminate combustible gas monitoring.

27. Perform MPC Lid-to-Shell weld pressure testing as follows:

ALARA Note:

Testing is performed before the MPC is drained for ALARA reasons. A weld repair is a lower dose activity if water remains inside the MPC.

a. *If performing a hydrostatic test*, attach the drain line to the vent port and route the drain line to the spent fuel pool or the plant liquid radwaste system *and connect the pressurized water supply to the drain port. If performing a pneumatic test, attach the pressure supply and vent line to the vent port and route the vent line to a suitable radwaste connection*.

ALARA Warning:

Water flowing from the MPC may carry activated particles and fuel particles. Apply appropriate ALARA practices around the drain line.

b. If performing a hydrostatic test, fill the MPC with either spent fuel pool water or plant demineralized water until water is observed flowing out of the vent port drain hose. *Refer to Tables 2.1.18 and 2.1.19 for required soluble boron concentration requirements.*

28. Perform a pressure test of the MPC as follows:

a. Close the drain valve and pressurize the MPC to the requirements given in Table 2.0.1.

b. Close the inlet valve and monitor the pressure for a minimum of 10 minutes. *The pressure shall not drop below the minimum test pressure during the performance of the test.*

c. Following the 10-minute hold period, visually examine the MPC lid-to-shell weld for leakage of water *(hydrostatic test) or helium using a bubble test solution (pneumatic test)*. The acceptance criterion is no observable leakage.

29. Release the MPC internal pressure, disconnect the inlet line and drain line from the vent and drain port RVOAs leaving the vent and drain port caps open.

30. Perform Required NDE inspections on MPC Lid to Shell Weld.

31. Repair any weld defects in accordance with the site's approved weld repair procedures. Reperform the Ultrasonic and Pressure tests if weld repair is performed.

32. Drain the MPC as follows:

ALARA Warning:
Dose rates will rise as water is drained from the MPC. Continuous dose rate monitoring is recommended.

a. Attach a regulated helium or nitrogen supply to the vent port.

b. Attach a drain line to the drain port.

c. Verify the correct pressure on the gas supply.

d. Open the gas supply valve and record the time at the start of MPC draindown.

e. Start the warming device, if used.

f. Blow the water out of the MPC until water ceases to flow out of the drain line. Shut the gas supply valve.

g. Disconnect the gas supply line from the MPC.

h. Disconnect the drain line from the MPC.

33. Vacuum Dry the MPC as follows:

Note:
Vacuum drying is performed to remove moisture and oxidizing gasses from the MPC. TThe vacuum drying process reduces the MPC internal pressure in stages. Dropping the internal pressure too quickly may cause the formation of ice in the fittings. Ice formation could result in incomplete removal of moisture from the MPC.

a. Attach the Vacuum Drying System (VDS) to the vent and drain port RVOAs.

Note:
The Vacuum Drying System may be configured with an optional fore-line condenser

Note:
To prevent freezing of water, the MPC internal pressure should be lowered in incremental steps. The Vacuum Drying System pressure will remain at about 30 torr until most of the liquid water has been removed from the MPC.

b. Open the VDS suction valve and reduce the MPC pressure to below 3 torr.

c. Shut the VDS valves and verify a stable MPC pressure on the vacuum gage.

Note:
The MPC pressure may rise due to the presence of water in the MPC. The dryness test may need to be repeated several times until all the water has been removed. Leaks in the Vacuum Drying System, damage to the vacuum pump, and improper vacuum gauge calibration may cause repeated failure of the dryness verification test. These conditions should be checked as part of the corrective actions if repeated failure of the dryness verification test is occurring.

d. Perform the MPC dryness verification test in accordance with the acceptance criteria in the Technical Specification.

e. Close the vent and drain port valves.

f. Disconnect the VDS from the MPC.

g. Stop the warming device, if used.

h. Close the drain port RVOA cap and remove the drain port RVOA.

Note:
Helium backfill requires 99.995% (minimum) purity.

34. Backfill the MPC as follows:

a. Set the helium bottle regulator pressure to the appropriate pressure.

b. Purge the Helium Backfill System to remove oxygen from the lines.

c. Attach the Helium Backfill System (HBS) to the vent port and open the vent port.

d. Slowly open the helium supply valve while monitoring the pressure rise in the MPC.

Note:
If helium bottles need to be replaced, the bottle valve needs to be closed and the entire regulator assembly transferred to the new bottle.

e. Carefully backfill the MPC to greater than 0 psig (0 kpa) and less than the maximum pressure in the Technical Specification.

f. Disconnect the HBS from the MPC.

g. Close the vent port RVOA and disconnect the vent port RVOA.

35. Dry and Backfill the MPC as follows (FHD Method):

Note:
Helium backfill shall be in accordance with the Technical Specification using 99.995% (minimum) purity. When using the FHD system to perform the MPC helium backfill, the FHD system shall be evacuated or purged and the system operated with 99.995% (minimum) purity helium.
Note:
MPC internal pressure during FHD operation must be $\leq$ 75 psia (517 kPa), to comply with Technical Specification.
Caution:
In the event of an FHD System failure, the MPC cavity pressure must be set to at least 72 psig (496 kPa) to place the MPC in an acceptable condition.

a. Attach the moisture removal system to the vent and drain port RVOAs. Other equipment configurations that achieve the same results may also be used.

b. Circulate the drying gas through the MPC while monitoring the circulating gas for moisture. Collect and remove the moisture from the system as necessary.

c. Continue the monitoring and moisture removal until Technical Specification limit is met for MPC dryness.

d. Continue operation of the FHD system with the demoisturizer on.

e. While monitoring the temperatures into and out of the MPC, adjust the helium pressure in the MPC to provide a fill pressure as required by the Technical Specifications.

f. Open the FHD bypass line.

g. Close the vent and drain port RVOAs.

h. *Shutdown the FHD system and disconnect it from the RVOAs.*

i. *Remove the vent and drain port RVOAs.*

36. Weld the vent and drain port cover plates as follows:

    a. Wipe the inside area of the vent and drain port recesses to dry and clean the surfaces.

    b. Place the cover plate over the vent port recess.

    c. Insert the nozzle of the helium supply into the vent port recess to displace the oxygen.

Note:
Helium gas is required to be injected into the port recesses to ensure that the leakage test is valid.

    d. Weld the cover plate and perform NDE with approved procedures. (See 9.1 and Table 2.2.15)

Note:
NDE personnel shall be qualified per the requirements of Section V of the ASME Code [8.1.3] or a site-specific program.

    e. Repeat Steps 36.a through 36.d for the drain port cover plate.

37. Perform a leakage test of the MPC vent and drain port cover plates as follows:

Note:
The leakage detector may detect residual helium in the atmosphere from the helium injection process. If the leakage tests detects a leak, the area should be blown clear with compressed air or nitrogen and the location should be retested.

    a. Flush the area around the vent and drain cover plates with compressed air or nitrogen to remove any residual helium gas.

    b. Perform a helium leakage rate test of vent and drain cover plate welds in accordance with the Mass Spectrometer Leak Detector (MSLD) manufacturer's instructions and ANSI N14.5 [8.1.2]. The MPC helium leakage rate test acceptance criteria are specified in the Technical Specification.

    c. Repair any weld defects in accordance with the site's approved code weld repair procedures. Reperform the leakage test as required.

38. Weld the MPC closure ring as follows:

ALARA Note:
The closure ring is installed by hand. No tools are required.

a. Install and align the closure ring.

b. Weld the closure ring to the MPC shell and the MPC lid, and perform NDE with approved procedures (See 9.1 and Table 2.2.15).

c. Remove the Automated Welding System (if used).

d. If necessary, remove the AWS baseplate shield. See Figure 8.1.8 for rigging.

8.1.7 Preparation for Storage

1. Remove the annulus shield and seal surface protector and store it in an approved plant storage location

ALARA Warning:
Dose rates will rise around the top of the annulus as water is drained from the annulus. Apply appropriate ALARA practices.

2. Attach a drain line to the HI-STAR 100 overpack drain connector and drain the remaining water from the annulus to the spent fuel pool or the plant liquid radwaste system.

3. Install the overpack closure plate as follows:

   a. Remove any waterproof tape or bolt plugs used for contamination mitigation.

   b. Clean the closure plate seal seating surface and the HI-STAR 100 overpack seal seating surface and install new overpack closure plate mechanical seals.

   c. Remove the test port plug and store it in a site-approved location. Discard any used metallic seals.

   d. Install the closure plate. Disconnect the closure plate lifting eyes and install the bolt hole plugs in the empty bolt holes (See Table 8.1.3 for torque requirements).

   e. Install and torque the closure plate bolts. See Table 8.1.3 for torque requirements.

   f. Remove the vent port cover plate and remove the port plug and seal. Discard any used mechanical seals.

4. Dry the overpack annulus as follows:

   a. Disconnect the drain connector from the overpack.

   b. Install the drain port plug with a new seal and torque the plug. See Table 8.1.3 for torque requirements. Discard any used metallic seals.

Note:
Preliminary annulus vacuum drying may be performed using the test cover to improve flow rates and reduce vacuum drying time. Dryness testing and helium backfill shall use the backfill tool.

c. Load the backfill tool with the HI-STAR 100 overpack vent port plug and the vent port with a new plug seal. Attach the backfill tool to the HI-STAR 100 overpack vent port with the plug removed. See Table 8.1.3 for torque requirements.

Note:
To prevent freezing of water, the MPC internal pressure should be lowered in incremental steps. The Vacuum Drying System pressure will remain at about 30 torr (4 kPa) until most of the liquid water has been removed from the overpack.

d. Open the Vacuum Drying System suction valve and reduce the HI-STAR 100 overpack pressure to below 3 torr (0.4 kPa).

Note:
The annulus pressure may rise due to the presence of water in the HI-STAR 100 overpack. The dryness test may need to be repeated several times until all the water has been removed. Leaks in the Vacuum Drying System, damage to the vacuum pump, and improper vacuum gauge calibration may cause repeated failure of the dryness verification test. These conditions should be checked as part of the corrective actions if repeated failure of the dryness verification test is occurring.

e. Perform a HI-STAR 100 overpack Annulus Dryness Verification in accordance with the Technical Specifications

5. Backfill, and leakage test the overpack as follows:

a. Attach the helium supply to the backfill tool.

b. Verify the correct pressure on the helium supply (pressure set to10+4/-0 psig (69 + 28 /-0 kPa) and open the helium supply valve.

c. Backfill the HI-STAR 100 overpack annulus in accordance with the Technical Specification.

d. Install the overpack vent port plug and torque. See Table 8.1.3 for torque requirements.

e. Disconnect the overpack backfill tool from the vent port.

f. Flush the overpack vent port recess with compressed air to remove any standing helium gas.

g. Install the overpack test cover to the overpack vent port. See Table 8.1.3 for torque requirements.

h. Evacuate the test cavity per the MSLD manufacturer's instructions and isolate the vacuum pump from the overpack test cover.

i. Perform a leakage rate test of overpack vent port plug per the MSLD manufacturer's instructions and ANSI N14.5 [8.1.2]. The helium leakage rate test acceptance criterion is specified in the Technical Specification.

j. Remove the overpack test cover and install a new metallic seal on the overpack vent port cover plate. Discard any used metallic seals.

k. Install the vent port cover plate and torque the bolts. See Table 8.1.3 for torque requirements.

l. Repeat Steps 5.f through 5.k for the overpack drain port.

6. Leak test the overpack closure plate inner mechanical seal as follows:

   a. Attach the closure plate test tool to the closure plate test port with the MSLD attached. See Table 8.1.3 for torque requirements.

   b. Evacuate the closure plate test port tool and closure plate inter-seal area per the MSLD manufacturer's instructions.

   c. Perform a leakage rate test of overpack closure plate inner mechanical seal in accordance with the MSLD manufacturer's instructions and ANSI N14.5 [8.1.2]. The helium leakage rate test acceptance criterion is specified in the Technical Specification.

   d. Remove the closure plate test tool from the test port and install the test port plug with a new mechanical seal (see Table 8.1.3).

7. Drain the Temporary Shield Ring, if used, and place in an approved plant storage location.

ALARA Warning:
For ALARA reasons, decontamination of the overpack bottom shall be performed using a remotely operated cleaning device. If the overpack is to be down-ended on the transport frame, the bottom shield should be installed quickly. Personnel should remain clear of the bottom of the overpack.

8. Raise the HI-STAR 100 overpack and decontaminate the overpack bottom and perform a final survey and decontamination of the overpack. The acceptance criteria are the user's site requirements for transporting items out of the radiological controlled area or the Technical Specifications (whichever is more restrictive).

9. Verify that the HI-STAR 100 overpack dose rates are within the requirements of Technical Specifications.

8.1.8 Placement of the HI-STAR 100 Overpack into Storage

1. Secure the HI-STAR 100 overpack to the transporter as necessary.

2. Verify lifting requirements of the Technical Specifications are met.

3. Remove the transporter wheel chocks (if necessary) and transfer the HI-STAR 100 overpack to the ISFSI along the site-approved transfer route.

Note:
The HI-STAR 100 minimum pitch shall be 12 feet (3.66 m) (nominal).

4. Transfer the HI-STAR 100 overpack to its designated storage location at the appropriate pitch.

5. Install the HI-STAR 100 overpack pocket trunnion plugs and shear ring segments, if necessary. See Table 8.1.3 for reference torque values.

6. If used, install the Overpack Bottom Ring. (The optional overpack bottom ring is used to reduce dose rates around the base of the HI-STAR 100 overpack.)

Table 8.1.1

ESTIMATED HANDLING WEIGHTS OF HI-STAR 100 SYSTEM COMPONENTS††††

Component	Weight, lbs (kg)			Case† Applicability			
	MPC-24	MPC-68	MPC-32	1	2	3	4
Empty HI-STAR 100 overpack (without closure plate)	145,726	145,726	145,726 (66,100)	1	1	1	1
HI-STAR 100 overpack lid (closure plate without rigging)	7,984	7,984	7,984 (3,621)		1	1	1
Empty MPC (without lid or closure ring)	29,075	28,502	24,453 (11,092)	1	1	1	1
MPC lid (without fuel spacers or drain line)	9677	10,194	9,677 (4,389)	1	1	1	1
MPC Closure Ring	145	145	145 (66)		1	1	1
MPC Lower Fuel Spacers (variable)††	401	258	0 (0)	1	1	1	1
MPC Upper Fuel Spacers (variable)††	144	315	0 (0)	1	1	1	1
MPC Drain Line	50	50	50 (23)	1	1	1	1
Fuel (design basis without non-fuel bearing components)	36,360	42,092	55,040 (24,966)	1	1	1	1
Damaged Fuel Container (Dresden 1)	0	150	0 (0)				
Damaged Fuel Container (Humboldt Bay)	0	120	0 (0)				
MPC water (with fuel in MPC)†††	17,630	16,957	17,630 (7,997)	1			
Annulus Water	280	280	280 (127)	1			
HI-STAR 100 overpack Lift Yoke (with slings)	3600	3600	3600 (1,633)	1	1		
Annulus Seal	50	50	50 (23)	1			
Lid Retention System (optional)	2300	2300	2300 (1,043)				
Transport Frame	6700	6700	6700 (3,039)				1
Overpack Bottom Cover (optional)	6400	6400	6400 (2,903)				1
Temporary Shield Ring (optional)	2500	2500	2500 (1,134)				
Automated Welding System Baseplate Shield (optional)	2000	2000	2000 (907)				
Automated Welding System Robot	1900	1900	145,726 (66,100)				
Pocket Trunnion Plugs (optional)	60	60	7,984 (3,621)			1	
Overpack Bottom Ring (optional)	1300	1300	24,453 (11,092)			1	

† See Table 8.1.2.

†† The fuel spacers referenced in this table are for the heaviest fuel assembly for each MPC. This yields the maximum weight of fuel assemblies and spacers. For MPC-32, the weight of the fuel spacers are less than the weight of the design basis fuel assembly and therefore not included in the maximum handling weight calculations.

††† Varies by fuel type and loading configuration. Users may opt to pump some water from the MPC prior to removal from the spent fuel pool to reduce the overall lifted weight.

†††† Actual component weights are dependent upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

TABLE 8.1.2
ESTIMATED HANDLING WEIGHTS
HI-STAR 100 OVERPACK†

Case No.	Load Handling Evolution	Weight,lbs (kg)		
		MPC-24	MPC-68	MPC-32
1	Loaded HI-STAR 100 Overpack Removal from Spent Fuel Pool	242,993	248,024	256,456 (116,326)
2	Loaded HI-STAR 100 Overpack Movement to transport device	233,162	238,866	246,625 (111,867)
3	Loaded HI-STAR 100 Overpack in Storage	230,922	236,626	244,385 (110,851)
4	Loaded HI-STAR 100 on Transport Frame During On-Site Handling	242,662	248,366	256,125 (116,176)

† See footnote †††† with Table 8.1.1

Note: The weight of the fuel spacers and the damaged fuel container are less than the weight of the design basis fuel assembly for each MPC and are therefore not included in the maximum handling weight calculations.

Table 8.1.3
HI-STAR 100 SYSTEM TORQUE REQUIREMENTS

Fastener	Torque,ft-lbs (N-m)	Pattern
Overpack Closure Plate Bolts†, ††	First Pass – Hand Tight Second Pass – Wrench Tight Third Pass – 700 +50/-50 (949 ± 68) Fourth Pass – 1400 +100/-100 (1898 ± 136) Final Pass – 2000 +250/-0 (2712 +339/-0)	Figure 8.1.31
Overpack Vent and Drain Port Cover Plate Bolts††	12+2/-0 (16.3 + 2.7/-0)	X-pattern
Overpack Vent and Drain Port Plugs	45+5/-2 (61 + 6.8/-2.7)	None
Closure Plate Test Port Plug	45 +5/-2 (61 + 6.8/-2.7)	None
Backfill Tool Test Cover Bolts††	16+2/-0 (22 +2.7/-0)	X-pattern
Shear Ring Segment Bolts	22+2/-0 (30 +2.7/-0)	None
Overpack Bottom Cover Bolts	200+20/-0 (271 + 27/-0)	None
Pocket Trunnion Plugs	Hand Tight	None
Upper Fuel Spacers	Hand Tight	None
Threaded Inserts (all)	Hand Tight	None

† Detorquing shall be performed by turning the bolts counter-clockwise in 1/3 turn +/- 30 degrees increments per pass according to Figure 8.1.31 for three passes. The bolts may then be removed.

†† Bolts shall be cleaned and inspected for damage or excessive wear (replaced if necessary) and coated with a light layer of Fel-Pro Chemical Products, N-5000, Nuclear Grade Lubricant (or equivalent).

Table 8.1.4
HI-STAR 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Description
Annulus Overpressure System (optional)	Not Important To Safety	The Annulus Overpressure System is used for supplemental protection against spent fuel pool water contamination of the external MPC shell and baseplate surfaces by providing a slight annulus overpressure. The Annulus Overpressure System consists of the quick disconnects water reservoir, reservoir valve and annulus connector hoses. User is responsible for supplying demineralized water to the location of the Annulus Overpressure System.
Annulus Shield (optional)	Not Important To Safety	A shield that is placed at the top of the annulus to provide supplemental shielding to the operators performing cask loading and closure operations. Shield segments are installed by hand, no crane or tools required.
Automated Welding System (optional)	Not Important To Safety	Used for remote welding of the MPC lid, vent and drain port cover plates and the MPC closure ring. The AWS consists of the robot, wire feed system, torch system, weld power supply and gas lines.
AWS Baseplate Shield (optional)	Not Important To Safety	The AWS baseplate shield provides supplemental shielding to the operators during the cask closure operations.
Backfill Tool	Not Important to Safety	Used to dry, backfill the HI-STAR 100 annulus and install the HI-STAR 100 overpack vent and drain port plugs. The backfill tool uses the same bolts as the HI-STAR 100 overpack vent and drain cover plates.
Blowdown Supply System	Not Important To Safety	Gas hose with pressure gauge, regulator used for blowdown of the MPC.
Cask Transporter	Not Important To Safety	Used for handling of the HI-STAR 100 overpack cask around the site. The cask transporter may take the form of heavy haul transfer trailer, special transporter or other equipment specifically designed for such function.
Closure Plate Test Tool	Not Important To Safety	Used to helium leakage test the HI-STAR 100 overpack Closure Plate inner mechanical seal.
Cool-Down System	Not Important To Safety	The Cool-Down System is a closed-loop forced ventilation cooling system used to gas-cool the MPC fuel assemblies down to a temperature water can be introduced without the risk of thermally shocking the fuel assemblies or flashing the water, causing uncontrolled pressure transients. The Cool-Down System is attached between the MPC drain and vent ports The CDS consists of the piping, blower, heat exchanger, valves, instrumentation, and connectors. The CDS is used only for unloading operations.
Drain Connector	Not Important To Safety	Used for draining the annulus water following cask closure operations. The Drain Connector consists of the connector pipe valve, and quick disconnect for adapting to the Annulus Overpressure System.
Four Legged Sling and Lifting Rings	Not Important To Safety (controlled under the user's rigging equipment program)	Used for rigging the HI-STAR 100 overpack upper shield lid, MPC lid, AWS Baseplate shield, and Automated Welding System Baseplate Shield. Consists of a four legged sling, lifting rings, shackles and a main lift link.
Forced Helium Dehydration	*Important to Safety Category B*	*Used for removal of residual moisture from the MPC following water draining. Calibration of the instrumentation used to confirm Tech Spec compliance shall be performed in accordance with the requirements*

Table 8.1.4
HI-STAR 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Description
System (FHD)		*for Important to Safety Category B, the remaining components of the system are NITS*
Helium Backfill System	Not Important To Safety	Used for helium backfilling of the MPC. System consists of the gas lines, mass flow monitor, integrator, and valved quick disconnect.
Hydrostatic Test System	Not Important To Safety	Used to hydrostatically test the MPC primary welds. The hydrostatic test system consists of the gauges, piping, pressure protection system piping and connectors.
Inflatable Annulus Seal	Not Important To Safety	Used to prevent spent fuel pool water from contaminating the external MPC shell and baseplate surfaces during in-pool operations.
Lid Retention System (optional)	User designated	The Lid Retention System provides three functions; it guides the MPC lid into place during underwater installation, establishes lift yoke alignment with the HI-STAR 100 overpack trunnions, and locks the MPC lid in place during cask handling operations between the pool and decontamination pad. The device consists of the retention disk, alignment pins, lift yoke connector links and lift yoke attachment bolts.
Lift Yoke	User designated	Used for HI-STAR 100 overpack cask handling when used in conjunction with the overhead crane. The lift yoke consists of the lift yoke assembly and crane hook engagement pin(s). The lift yoke is a modular design that allows inspection, disassembly, maintenance and replacement of components.
MPC Upending Frame	Not Important to Safety	A steel frame used to evenly support the MPC during upending operations.
MSLD (Helium Leakage Detector)	Not Important To Safety	Used for helium leakage testing of the MPC vent/drain port welds.
Overpack Bottom Cover (optional)	Not Important to Safety	A cup-shaped shield used to reduce dose rates around the HI-STAR 100 overpack bottom end when operated in the horizontal orientation.
Overpack Bottom Ring (optional)	Not Important to Safety	Segmented shield ring that fits under the HI-STAR 100 overpack neutron shield. Used to reduce dose rates around the HI-STAR 100 overpack bottom end.
Overpack Test Cover	Not Important to Safety	Used to helium leakage test the HI-STAR 100 overpack vent and drain port plug seals.
Seal Surface Protector (optional)	Not Important to Safety	Used to protect the HI-STAR 100 overpack mechanical seal seating surface during loading and MPC closure operations.
Temporary Shield Ring (optional)	Not Important To Safety	Fits on the cask neutron shield around the upper forging and provides supplemental shielding to personnel performing cask loading and closure operations.
Threaded Inserts	Not Important To Safety	Used to fill the empty threaded holes in the HI-STAR 100 overpack and MPC.
Transport Frame (optional)	Not Important To Safety	A frame used to support the HI-STAR 100 overpack during on-site movement and upending/downending operations. The frame consists of the rotation trunnions, main frame beams and front saddle and lift points.
Transport Cradle and Cask Tilting Plate (optional)	Not Important to Safety	A cradle used to support the HI-STAR 100 overpack during on-site movement and upending/downending operations. The cradle consists of a cask tilting plate that is bolted to the bottom of the HI-STAR 100 (when pocket trunnions are not used), main frame beams and front saddle and lift points. .
Vacuum Drying	Not Important To Safety	Used for removal of residual moisture from the MPC and HI-STAR 100 Overpack annulus following water

Table 8.1.4
HI-STAR 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Description
System		draining. Used for evacuation of the MPC to support backfilling operations. Used to support test volume samples for MPC unloading operations. The VDS consists of the vacuum pump, piping, skid, gauges, valves, inlet filter, flexible hoses, connectors, control system.
Vent and Drain RVOAs	Not Important To Safety	Used to drain, dry, inert and fill the MPC through the vent and drain ports. The vent and drain RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operation.
Weld Removal System	Not Important To Safety	Semi-automated weld removal system used for removal of the MPC to shell weld, MPC to closure ring weld and closure ring to MPC shell weld. The WRS mechanically removes the welds using a high-speed cutter.

Table 8.1.5
HI-STAR 100 SYSTEM INSTRUMENTATION SUMMARY FOR LOADING AND UNLOADING OPERATIONS†

Instrument	Function
Dose Rate Monitors/Survey Equipment	Monitors dose rate and contamination levels and ensures proper function of shielding. Ensures assembly debris is not inadvertently removed from the spent fuel pool during overpack removal.
Flow Rate Monitor	Monitors the air flow rate during assembly cool-down.
Helium Mass Flow Monitor (optional)	Determines the amount of helium introduced into the MPC during backfilling operations. Includes integrator.
Helium Mass Spectrometer Leak Detector (MSLD)	Ensures leakage rates of welds are within acceptance criteria.
Helium Pressure Gauges	Ensures correct helium backfill pressure during backfilling operation.
Volumetric Testing Rig	Used to assess the integrity of the MPC lid-to-shell weld.
Pressure Gauge	Ensures correct helium pressure during fuel cool-down operations.
Hydrostatic Test Pressure Gauge	Used for hydrostatic testing of MPC lid-to-shell weld.
Temperature Gauge	Monitors the state of fuel cool-down prior to MPC flooding.
Temperature Probe	For fuel cool-down operations
Vacuum Gauges	Used for vacuum drying operations and to prepare an MPC evacuated sample bottle for MPC gas sampling for unloading operations.
Water Pressure Gauge	Used for performance of the MPC Hydrostatic Test.

Note: This table summarizes the instruments identified in the procedures for cask loading and unloading operations. Alternate instruments are acceptable as long as they can perform appropriate measurements.

† All safety significant instruments require calibration in accordance with Holtec's QA program. See figures at the end of this section for additional instruments, controllers and piping diagrams.

Table 8.1.6
HI-STAR 100 OVERPACK INSPECTION CHECKLIST

HI-STAR 100 Overpack Closure Plate:

1. Lifting rings shall be inspected for general condition and date of required load test certification.
2. The test port shall be inspected for dirt and debris, hole blockage, thread condition, presence or availability of the port plug and replacement mechanical seals.
3. The mechanical seal grooves shall be inspected for cleanliness, dents, scratches and gouges and the presence or availability of replacement mechanical seals.
4. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
5. All closure plate surfaces shall be relatively free of dents, scratches, gouges or other damage.
6. The vent port plug shall be inspected for thread condition, and sealing surface condition (scratches, gouges).
7. Overpack vent port shall be inspected for presence or availability of port plugs, hole blockage, plug seal seating surface condition.
8. Overpack vent port cover plate shall be inspected for cleanliness, scratches, dents, and gouges, availability of retention bolts, and of replacement mechanical seals.

HI-STAR 100 Overpack Main Body:

1. The impact limiter attachment bolt holes shall be inspected for dirt and debris and thread condition.
2. The mechanical seal seating surface shall be inspected for cleanliness, scratches, and dents or gouges.
3. The drain port plug shall be inspected for thread condition, and sealing surface condition (scratches, gouges).
4. The closure plate bolt holes shall be inspected for dirt, debris and thread damage.
5. Painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
6. Trunnions shall be inspected for deformation, cracks, thread damage, end plate damage, corrosion, excessive galling, damage to the locking plate, presence or availability of locking plate and end plate retention bolts.

Table 8.1.6
HI-STAR 100 OVERPACK INSPECTION CHECKLIST
(continued)

7. Pocket trunnion recesses, if provided, shall be inspected for indications of over stressing (i.e., cracks, deformation, excessive wear).
8. Overpack drain port cover plate shall be inspected for cleanliness, scratches, dents, and gouges, availability of retention bolts, availability of replacement mechanical seals.
9. Overpack drain port shall be inspected for presence or availability of port plug, availability of replacement mechanical seals, hole blockage, plug seal seating surface condition.
10. Annulus inflatable seal groove shall be inspected for cleanliness, scratches, dents, gouges, sharp corners, burrs or any other condition that may damage the inflatable seal.
11. The overpack rupture disks shall be inspected for presence or availability and the top surface of the disk shall be visually inspected for holes, cracks, tears or breakage.
12. The nameplate shall be inspected for presence and general condition.
13. The removable shear ring shall be inspected for fit and thread condition.

Table 8.1.7
MPC RECEIPT INSPECTION CHECKLIST

MPC Lid and Closure Ring:
1. The MPC lid and closure ring surfaces shall be relatively free of dents, gouges or other shipping damage.
2. The drain line shall be inspected for straightness, thread condition, and blockage.
3. Upper fuel spacers (if used) shall be inspected for availability and general condition. Plugs shall be available for non-used spacer locations.
4. Lower fuel spacers (if used) shall be inspected for availability and general condition.
5. Drain and vent port cover plates shall be inspected for availability and general condition.
6. Serial numbers shall be inspected for readability.

MPC Main Body:
1. All visible MPC body surfaces shall be inspected for dents, gouges or other shipping damage.
2. Fuel cell openings shall be inspected for debris, dents and general condition.
3. Lift lugs shall be inspected for general condition.
4. Verify proper MPC basket type for contents.
5. Inspect drain guide tube for debris, dents and general condition.

LOCATION: CASK RECEIVING AREA
REMOVE HI-STAR CLOSURE PLATE
INSTALL MPC
INSTALL UPPER FUEL SPACERS
INSTALL LOWER FUEL SPACERS
FILL MPC AND ANNULUS
INSTALL ANNULUS SEAL
PLACE HI-STAR IN SPENT FUEL POOL
LOCATION: SPENT FUEL POOL
LOAD FUEL ASSEMBLIES INTO MPC
PERFORM ASSEMBLY IDENTIFICATION VERIFICATION
INSTALL DRAIN LINE TO MPC LID
ALIGN MPC LID AND LIFT YOKE
INSTALL MPC LID
REMOVE HI-STAR FROM SPENT FUEL POOL
LOCATION: CASK PREPARATION AREA
DECONTAMINATE HI-STAR BOTTOM
SET HI-STAR IN CASK PREPARATION AREA
MEASURE DOSE RATES AT MPC LID
DECONTAMINATE HI-STAR AND LIFT YOKE
INSTALL TEMPORARY SHIELD RING
REMOVE INFLATABLE ANNULUS SEAL
LOWER ANNULUS WATER LEVEL SLIGHTLY
SMEAR MPC LID TOP SURFACES
INSTALL ANNULUS SHIELD
LOWER MPC WATER LEVEL
WELD MPC LID
PERFORM NDE ON MPC LID WELD
RAISE MPC WATER LEVEL

PERFORM PRESSURE TEST ON MPC
VACUUM DRY MPC (ALTERNATIVE FHD SYTEM MAY BE USED)
PERFORM MPC DRYNESS VERIFICATION TEST
BACKFILL MPC
WELD VENT AND DRAIN PORT COVER PLATES
PERFORM NDE ON COVER PLATE WELDS
PERFORM LEAKAGE TEST ON COVER PLATES
WELD MPC CLOSURE RING
PERFORM NDE ON CLOSURE RING WELDS
DRAIN ANNULUS
PERFORM SURVEYS ON HI-STAR
INSTALL HI-STAR CLOSURE PLATE
VACUUM DRY HI-STAR ANNULUS
BACKFILL HI-STAR ANNULUS
LEAKTEST HI-STAR ANNULUS
REMOVE TEMPORARY SHIELD RING
PERFORM FINAL SURVEYS ON HI-STAR
PERFORM SHIELDING EFFECTIVENESS TESTING ON HI-STAR
LOCATION: ISFSI
PLACE HI-STAR IN STORAGE
INSTALL HI-STAR POCKET TRUNNION PLUGS AND BOTTOM RING (OPTIONAL)

Figure 8.1.1; Loading Operations Flow Diagram

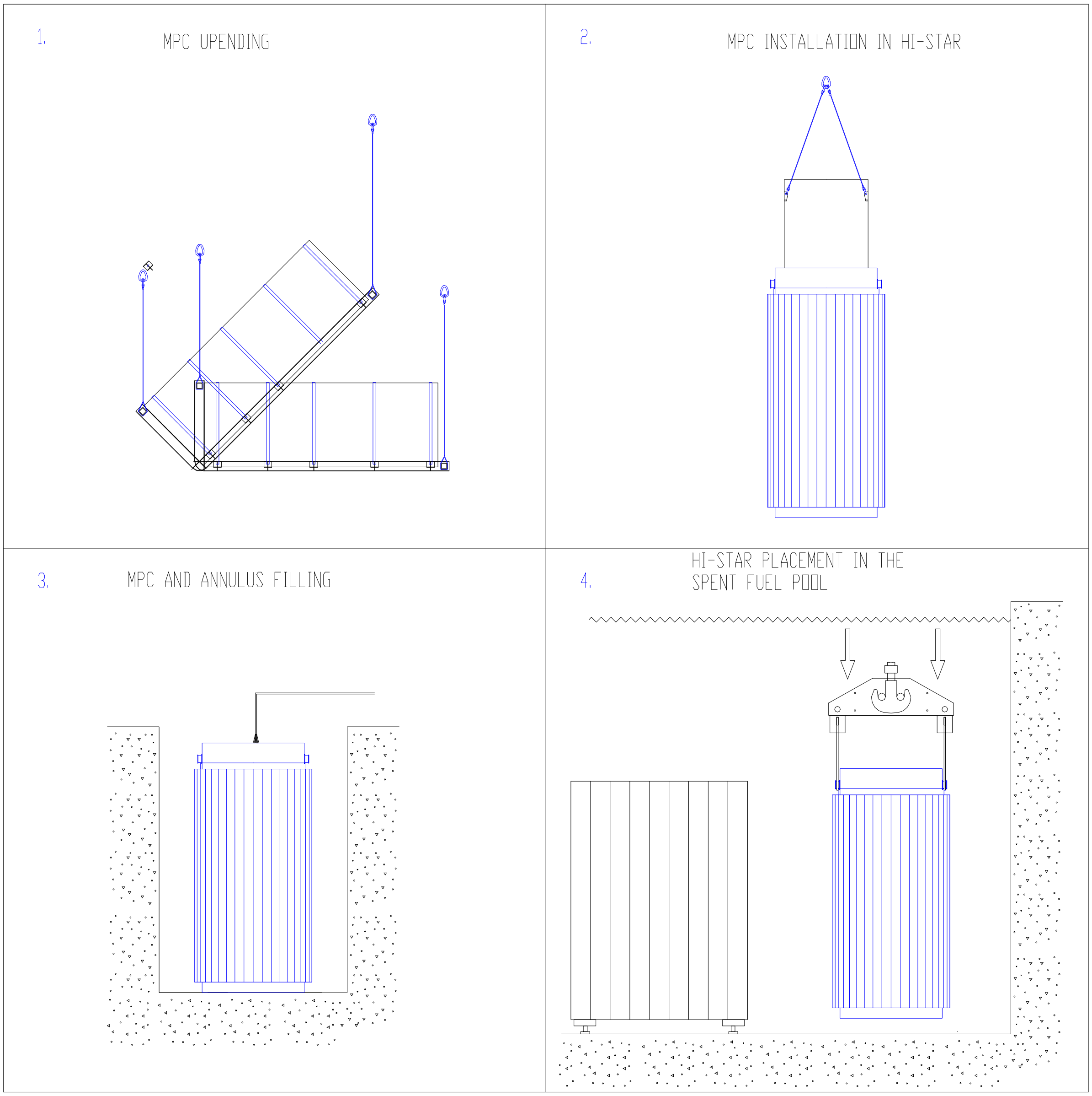

Figure 8.1.2a; Major HI-STAR 100 Loading Operations (Sheet 1 of 3)

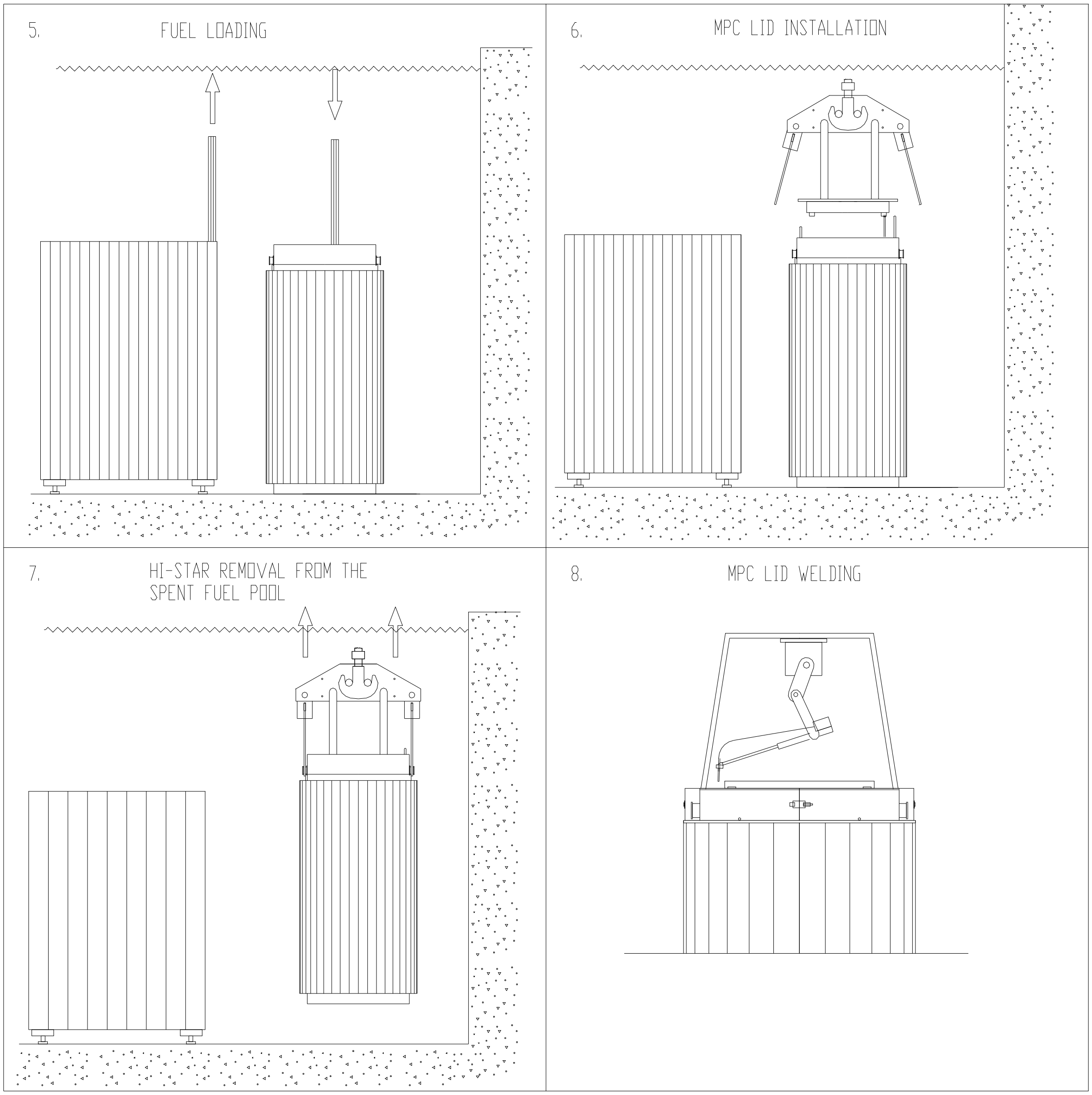

Figure 8.1.2b; Major HI-STAR 100 Loading Operations (Sheet 2 of 3)

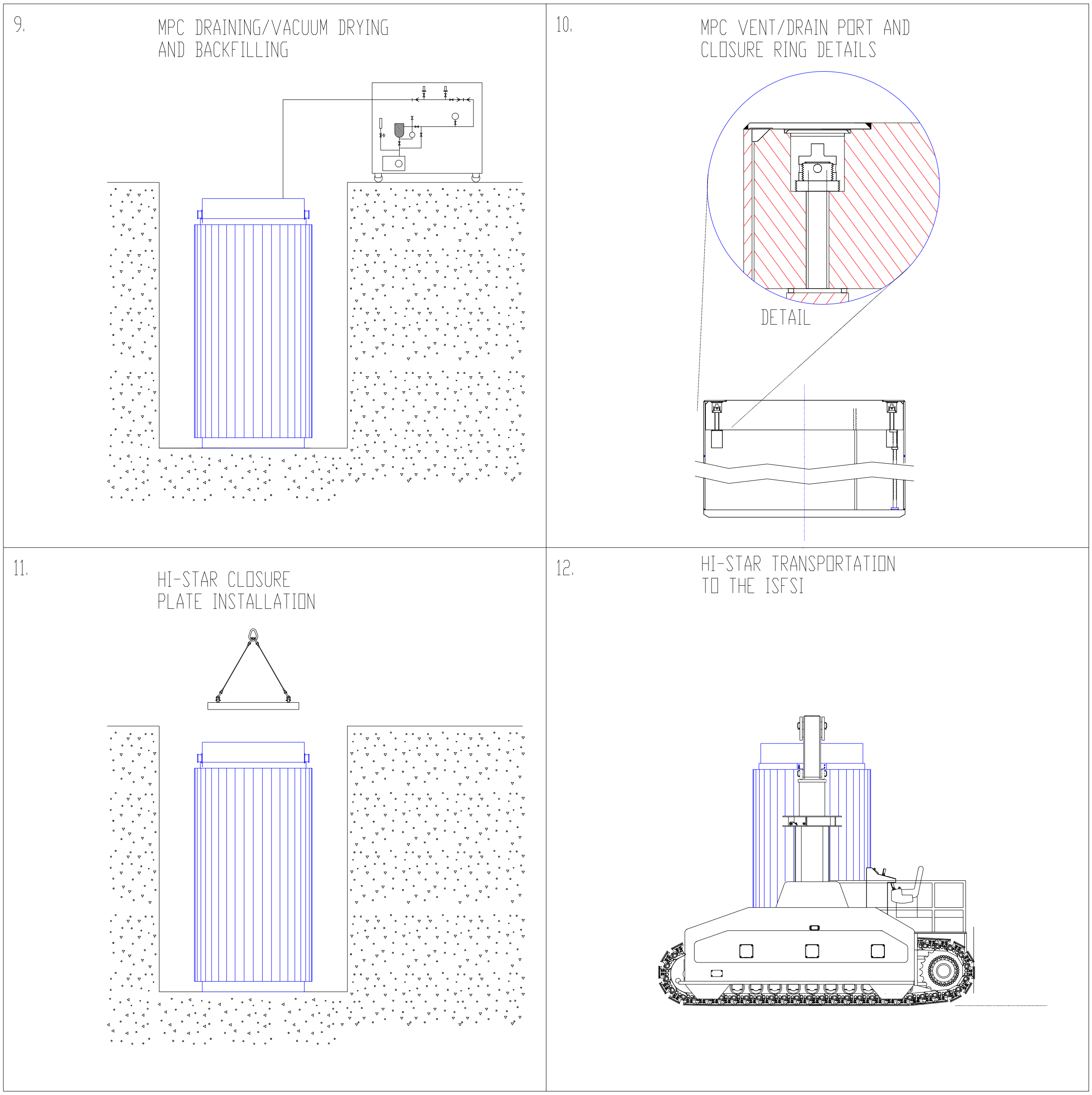

Figure 8.1.2c; Major HI-STAR 100 Loading Operations (Sheet 3 of 3)

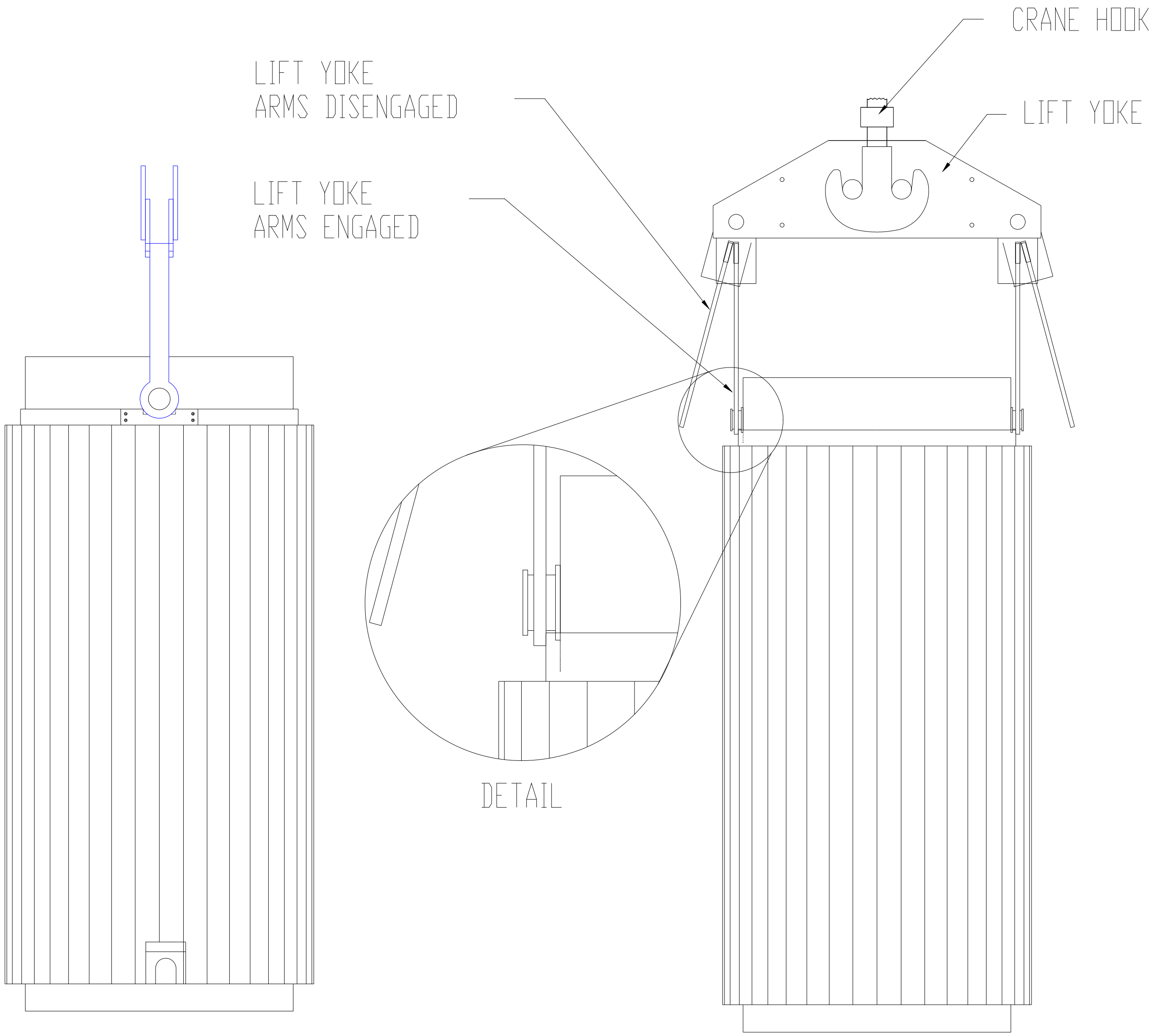

Figure 8.1.3; Lift Yoke Engagement and Vertical HI-STAR Handling

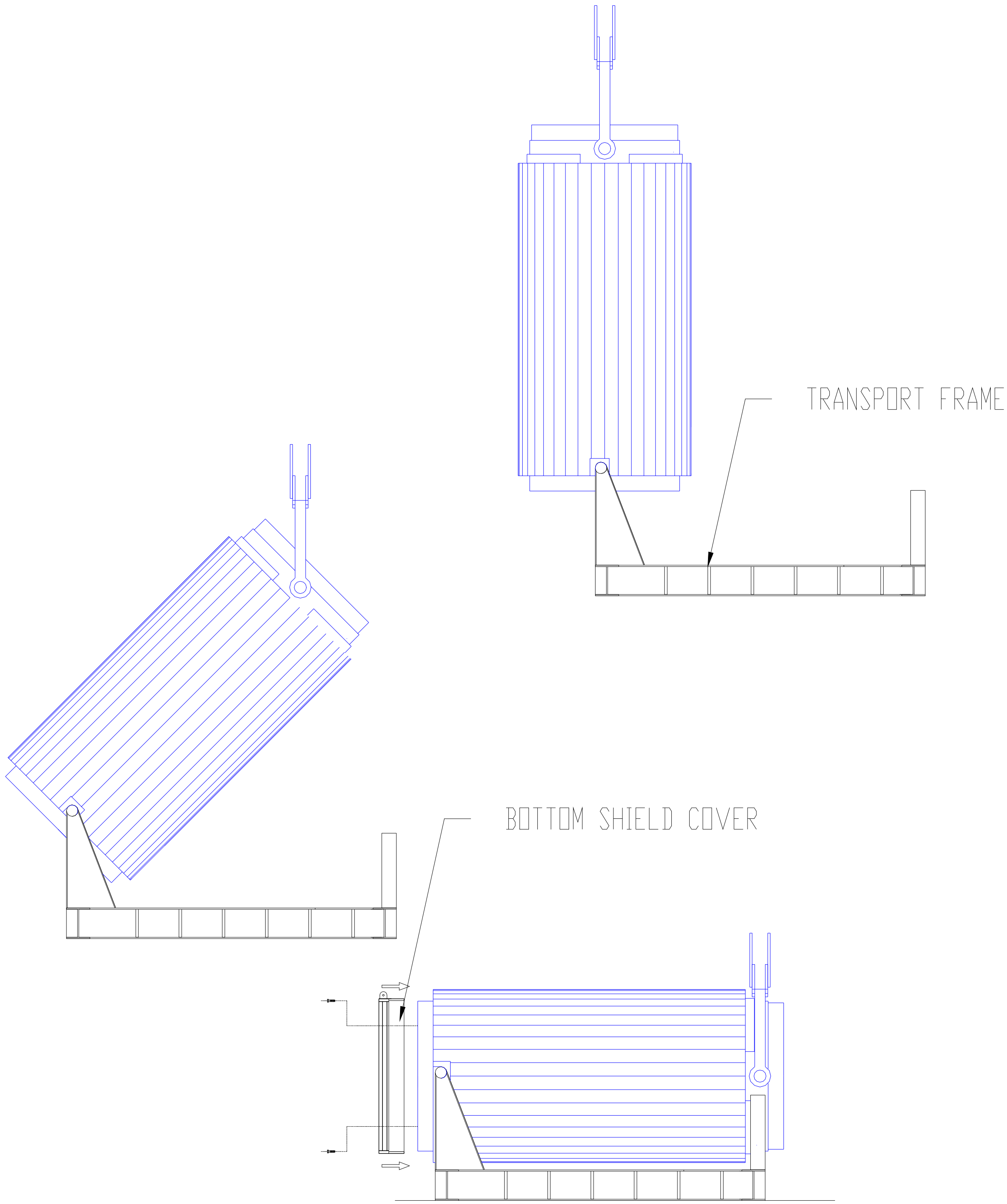

Figure 8.1.4; HI-STAR Upending/Downending in the Transport Frame

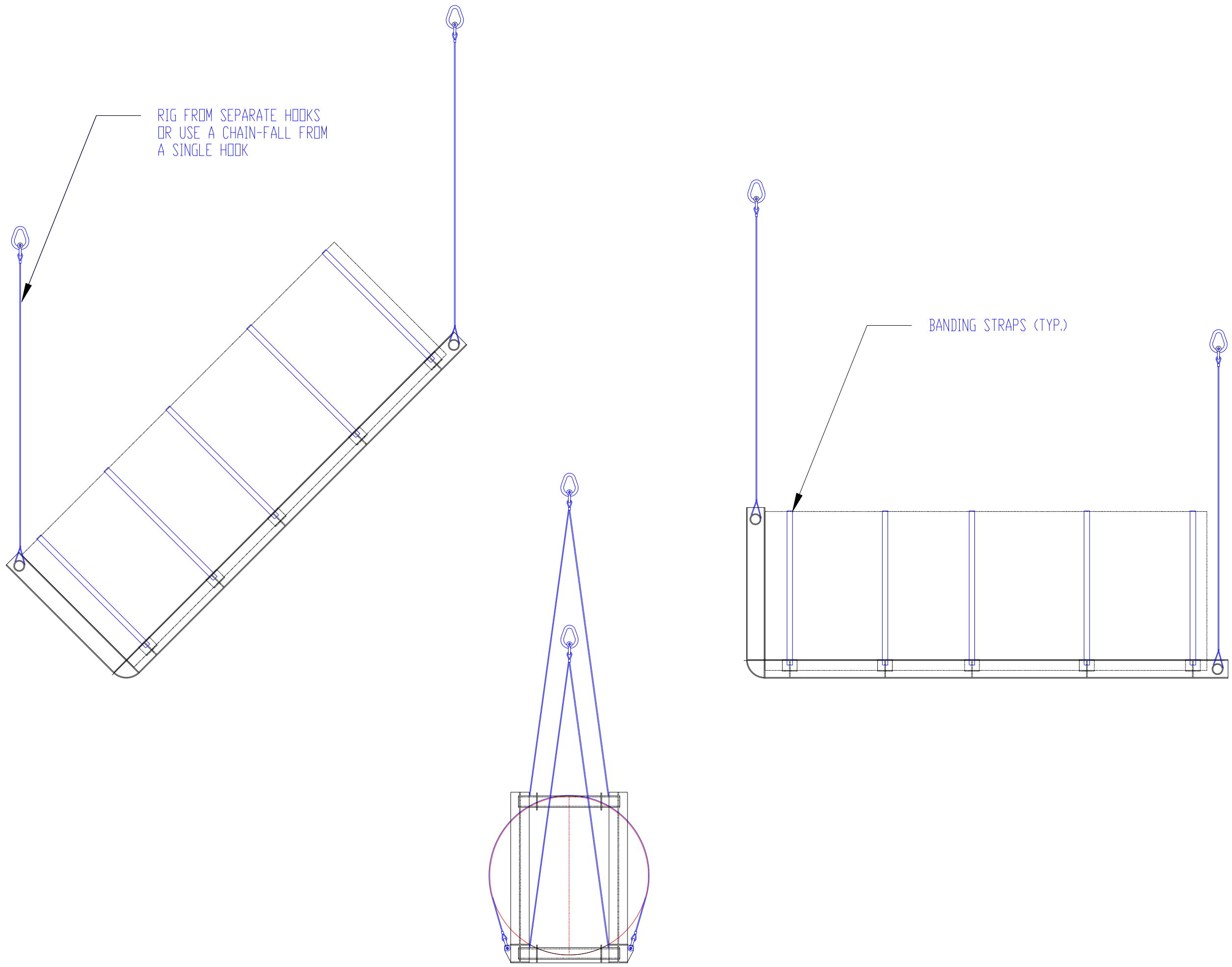

Figure 8.1.5; MPC Upending in the MPC Upending Frame

Figures 8.1.6 and 8.1.7 INTENTIONALLY DELETED

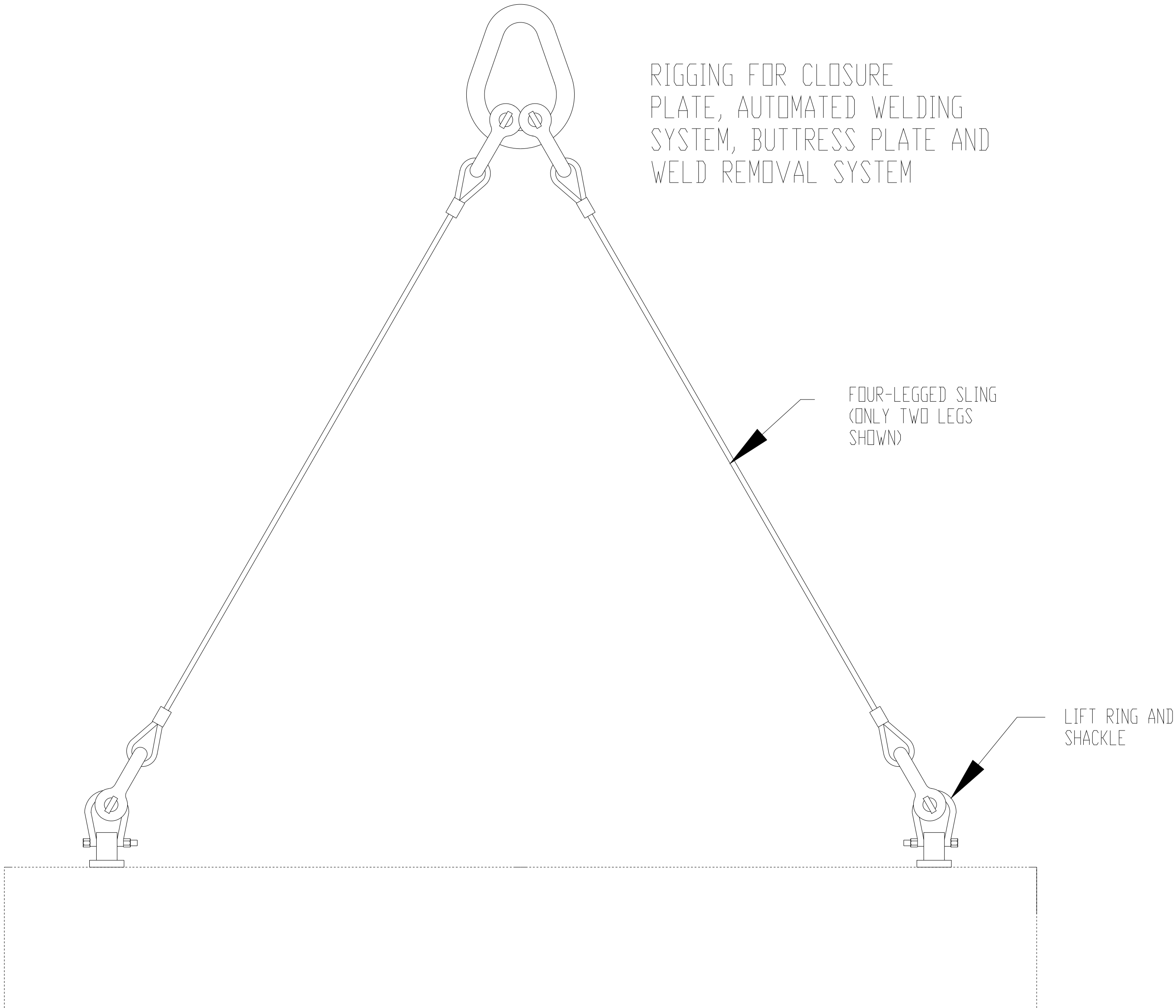

Figure 8.1.8; MPC Lid and HI-STAR Accessory Rigging

Figure 8.1.9 through 8.1.30; INTENTIONALLY DELETED

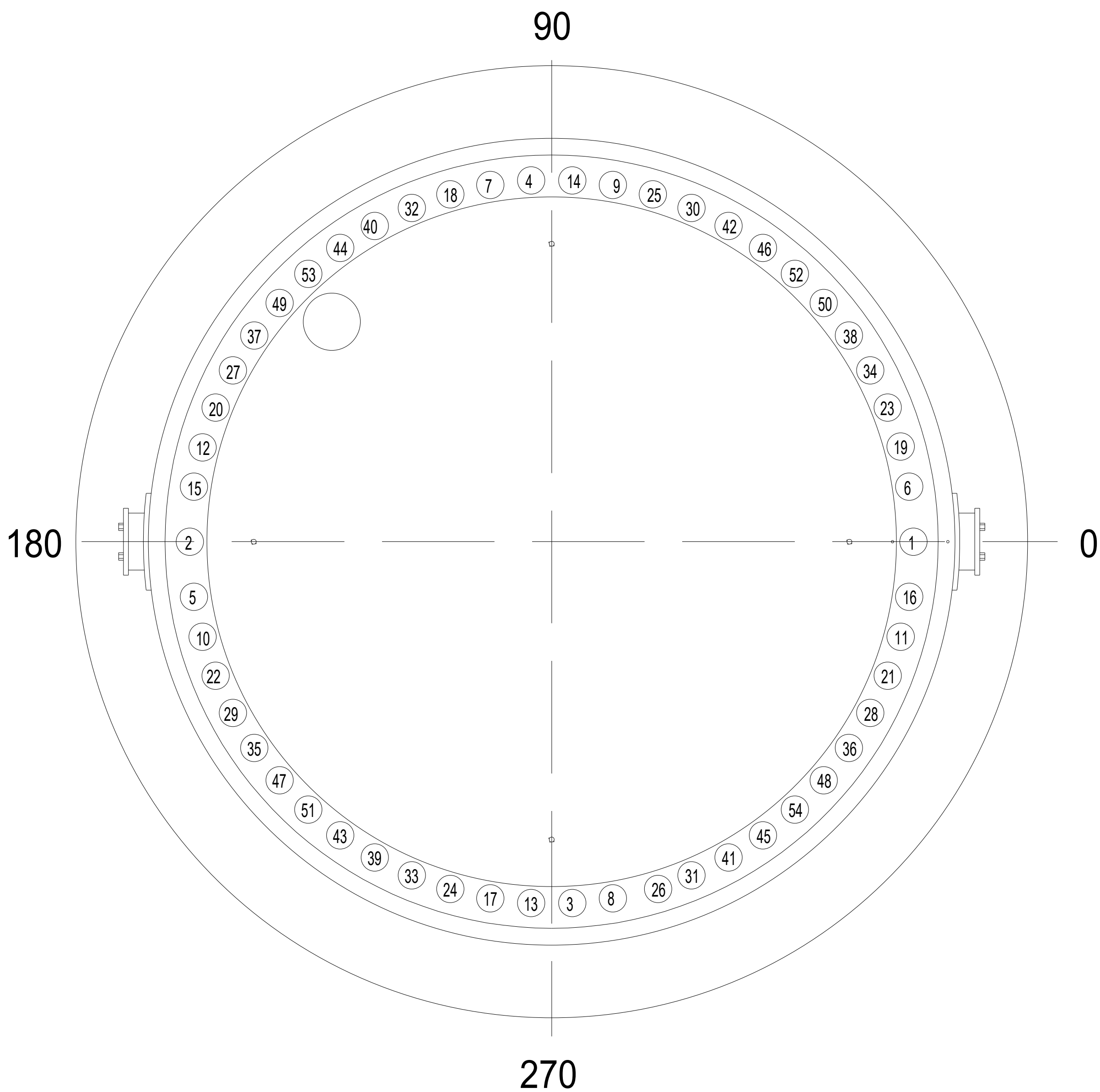

Figure 8.1.31; Recommended HI-STAR Closure Plate Bolt Torquing Pattern

8.2 ISFSI OPERATIONS

The HI-STAR 100 System is a totally passive system. Maintenance on the HI-STAR 100 system is typically limited to cleaning and touch-up painting of the HI-STAR 100 overpacks. In the event of significant damage to the HI-STAR 100 System, caused by an extenuating event such as an extreme environmental phenomenon, the cask may need to be unloaded and repaired If necessary, the procedures in Section 8.1 may be used to reposition a HI-STAR 100 overpack for minor repairs and maintenance. In extreme cases, Section 8.3 may be used as guidance for unloading the MPC from the HI-STAR 100 overpack. The procedures contained in the HI-STORM FSAR [8.2.1] may be used to transfer the MPC into a HI-STORM overpack or HI-STAR 100 overpack.

8.3 PROCEDURE FOR UNLOADING THE HI-STAR 100 SYSTEM IN THE SPENT FUEL POOL

8.3.1 Overview of HI-STAR 100 System Unloading Operations

The HI-STAR 100 System unloading procedures describe the general actions necessary to prepare the MPC for unloading, cool the stored fuel assemblies in the MPC, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover the HI-STAR 100 overpack and empty MPC. Special precautions are outlined to ensure personnel safety during the unloading operations, and to prevent the risk of MPC over-pressurization and thermal shock to the stored spent fuel assemblies. Figure 8.3.1 shows a flow diagram of the HI-STAR 100 overpack unloading operations. Figure 8.3.2 illustrates the major HI-STAR 100 overpack unloading operations.

ALARA Note:
The procedure described below uses the Weld Removal System, a remotely operated system that mechanically removes the welds. Users may opt to remove some or all of the welds using hand operated equipment. The decision should be based on dose rates, accessibility, degree of weld removal, and available tooling and equipment.

Refer to the boxes of Figure 8.3.2 for the following description. The HI-STAR 100 overpack is returned to the fuel building using any of the methodologies as described in Section 8.1 (Box 1). The HI-STAR 100 overpack vent port cover plate is removed and a gas sample is drawn from the HI-STAR 100 overpack annulus to determine the condition of the MPC confinement boundary. The annulus is depressurized and the HI-STAR 100 overpack closure plate is removed (Box 2). The Temporary Shield Ring is installed on the HI-STAR 100 overpack upper section. The Temporary Shield Ring and annulus are filled with plant demineralized water. The annulus and HI-TRAC top surfaces are protected from debris which will be produced when removing the MPC Lid. The MPC closure ring weld is removed using the Weld Removal System. The closure ring above the vent and drain ports and the vent and drain port cover plates are core-drilled and removed to access the vent and drain ports. (Box 3). The design of the vent and drain ports use metal-to-metal seals that prevent rapid decompression of the MPC and subsequent spread of contamination during unloading. The vent port RVOA is attached to the vent port and an evacuated sample bottle is connected. The vent port is opened slightly to allow the sample bottle to obtain a gas sample from inside the MPC. A gas sample is performed to assess the condition of the fuel assembly cladding. The MPC is cooled using a closed-loop heat exchanger to reduce the MPC internal temperature to allow water flooding (Box 4). The cool-down process gradually reduces the cladding temperature to a point where the MPC may be flooded with water without thermally shocking the fuel assemblies or causing uncontrolled pressure transients in the MPC from the formation of steam. Following the fuel cool-down, the MPC is filled with water at a specified rate (Box 5). The Weld Removal System then removes both the closure ring-to-MPC shell weld and the MPC lid to MPC shell welds. The Weld Removal System is removed with the MPC lid left in place (Box 6).

The top surfaces of the HI-STAR 100 overpack and MPC are cleared of metal shavings. The annulus shield is removed and the inflatable annulus seal is installed and pressurized. The MPC lid is rigged to the lift yoke or Lid Retention System and the lift yoke is engaged to the HI-STAR 100 overpack lifting trunnions. The HI-STAR 100 overpack is placed in the spent fuel pool and the MPC lid is removed (Box 7). All fuel assemblies are returned to the spent fuel storage racks (Box 8) and the MPC fuel cells are vacuumed to remove any assembly debris and crud. The HI-STAR 100 overpack and MPC are returned to the designated preparation area (Box 9) where the MPC water is pumped back into the spent fuel pool or liquid radwaste facility. The annulus water is drained and the MPC and overpack are decontaminated (Box 10 and 11).

8.3.2 HI-STAR 100 Overpack Recovery from Storage

1. Transfer the HI-STAR 100 overpack to the fuel building. The same methods may be used as was performed in the original cask placement operations. See Section 8.1.

2. Position the HI-STAR 100 overpack under the lifting device.

3. Place the HI-STAR 100 overpack in the designated preparation area.

ALARA Warning:
Gas sampling is performed to assess the condition of the MPC confinement boundary. If a leak is discovered in the MPC boundary, the user's Radiation Control organization may require special actions to vent the HI-STAR 100.

4. Perform annulus gas sampling as follows:

   a. Remove the overpack vent port cover plate and attach the backfill tool with a sample bottle attached. See Figure 8.3.3. Store the cover plate in a site-approved location.

   b. Using a vacuum pump, evacuate the sample bottle and backfill tool.

   c. Slowly open the vent port plug and gather a gas sample from the annulus. Reinstall the HI-STAR 100 overpack vent port plug.

   d. Evaluate the gas sample and determine the condition of the MPC confinement boundary.

5. If the confinement boundary is intact (i.e., no radioactive gas is measured) then vent the overpack annulus by removing the overpack vent port seal plug (using the backfill tool). Otherwise vent the annulus gas in accordance with instructions from Radiation Protection.

6. Remove the closure plate bolts. Store the closure plate bolts in a site-approved location.

7. Remove the overpack closure plate. Store the closure plate on cribbing to protect the seal seating surfaces.

8. Install the HI-STAR 100 overpack Seal Surface Protector.

Warning:
Annulus fill water may flash to steam due to high MPC shell temperatures. Water addition should be performed in a slow and controlled manner.

9. Remove the HI-STAR 100 overpack drain port cover and port plug and install the drain connector. Store the drain port cover plate and port plug in an approved storage location.

10. Slowly fill the annulus area with plant demineralized water to approximately 4 inches (10.2 cm) below the top of the MPC shell and install the annulus shield. Cover annulus & HI-TRAC top surfaces to protect them from debris produced when removing the MPC Lid.

11. Remove the MPC closure welds as follows:

ALARA Note:
The following procedures describe weld removal using the Weld Removal System. The Weld Removal System removes the welds with a high speed machine tool head. A vacuum head is attached to remove a majority of the metal shavings. Other methods of opening the MPC are acceptable. Weld removal may create an airborne radiation condition. Weld removal must be performed under the direction of the user's Radiation Protection organization.

    a. Install bolt plugs and/or waterproof tape on the closure plate bolt holes.

    b. Install the Weld Removal System on the MPC lid and core drill through the closure ring and vent and drain port cover plate welds.

12. Access the vent and drain ports.

ALARA Note:
The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage and withstand the long-term effects of temperature and radiation. The vent and drain port design prevents the need to hot tap into the penetrations during unloading operation and eliminate the risk of a pressurized release of gas from the MPC.

13. Take an MPC gas sample as follows:

    a. Attach the RVOA to the vent port.

    b. Attach a sample bottle to the vent port RVOA.

    c. Using the Vacuum Drying System, evacuate the RVOA and Sample Bottle.

    d. Slowly open the vent port cap using the RVOA and gather a gas sample from the MPC internal atmosphere.

    e. Close the vent port cap and disconnect the sample bottle.

ALARA Note:
The gas sample analysis is performed to determine the condition of the fuel cladding in the MPC. The gas sample may indicate that fuel with damaged cladding is present in the MPC. The results of the gas sample test may affect personnel protection and how the gas is processed during MPC depressurization.

f. Turn the sample bottle over to the site's Radiation Protection or Chemistry Department for analysis.

g. Install the RVOA in the drain port.

14. Perform Fuel Assembly Cool-Down as follows:

a. Deleted

b. Verify that the helium gas pressure regulator is set to the appropriate pressure.

c. Open the helium gas supply valve to purge the gas lines of air.

d. If necessary, slowly open the helium supply valve and increase the Cool-Down System pressure to MPC pressure. Close the helium supply valve.

e. Start the gas coolers.

f. Open the vent and drain port caps using the RVOAs.

g. Start the blower and monitor the gas exit temperature. Continue the fuel cool-down operations until the gas exit temperature meets the requirements of the Technical Specification.

Note:
Water filling should commence immediately after the completion of fuel cool-down operations to minimize fuel assembly heat-up. Prepare the water fill and vent lines in advance of water filling.

h. Prepare the MPC fill and vent lines. Route the vent port line several feet below the spent fuel pool surface or to the radwaste gas facility. Turn off the blower and disconnect the gas lines to the vent and drain port RVOAs. Attach the vent line to the MPC vent port and slowly open the vent line valve to depressurize the MPC.

i. Attach the water fill line to the MPC drain port and slowly open the water supply valve and establish a pressure less than 90 psi. Fill the MPC until bubbling from the vent line has terminated. Close the water supply valve on completion.

Caution:

Oxidation of neutron absorber panels contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during MPC cutting operations. The space below the MPC lid shall be exhausted or purged with inert gas prior to, and during MPC cutting operations to provide additional assurance that flammable gas concentrations will not develop in this space.

j. Disconnect both lines from the drain and vent ports and, connect a combustible gas monitor to the MPC vent port and check for combustible gas concentrations prior to and periodically during weld removal activities. Purge or evacuate the gas space under the lid as necessary.

k. Remove the closure ring-to-MPC shell weld and the MPC lid-to-shell weld using the Weld Removal System and remove the Weld Removal System.

l. Vacuum the top surfaces of the MPC and the HI-STAR 100 overpack to remove any metal shavings.

15. Install the inflatable annulus seal as follows:

a. Remove the annulus shield.

b. Manually insert the inflatable seal around the MPC.

c. Ensure that the seal is uniformly positioned in the annulus area.

d. Inflate the seal.

e. Visually inspect the seal to ensure that it is properly seated in the annulus. Deflate, adjust and inflate the seal as necessary.

16. Place HI-STAR 100 overpack in the spent fuel pool as follows:

a. Engage the lift yoke to the HI-STAR 100 overpack lifting trunnions, remove the MPC lid lifting threaded inserts and attach the MPC lid slings or Lid Retention System to the MPC lid.

b. If the Lid Retention System is used, inspect the lid bolts for general condition. Replace worn or damaged bolts with new bolts.

c. Install the Lid Retention System bolts if the Lid Retention System is used.

d. If used, fill the Annulus Overpressure System lines and reservoir with demineralized water and close the reservoir valve. Attach the Annulus Overpressure System to the HI-STAR 100 overpack.

Warning:
Cask placement in the spent fuel pool is the heaviest lift that occurs during the HI-STAR 100 unloading operations. Lifted loads must be within crane and heavy lifts limits. Users may elect to pump a measured quantity of water from the MPC prior to placement of the HI-STAR 100 in the spent fuel pool. See Table 8.1.1 and 8.1.2 for weight information.

e. Position the HI-STAR 100 overpack over the cask loading area with the basket aligned to the orientation of the spent fuel racks.

ALARA Note:
Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

f. Wet the surfaces of the HI-STAR 100 overpack and lift yoke with plant demineralized water while slowly lowering the HI-STAR 100 overpack into the spent fuel pool.

g. When the top of the HI-STAR 100 overpack reaches the approximate elevation of the reservoir, open the Annulus Overpressure System reservoir valve. Maintain the reservoir water level at approximately 3/4 full the entire time the cask is in the spent fuel pool.

h. If the Lid Retention System is used, remove the lid retention bolts when the top of the HI-STAR 100 overpack is accessible from the operating floor.

i. Place the HI-STAR 100 overpack on the floor of the cask loading area and disengage the lift yoke. Visually verify that the lift yoke is fully disengaged.

j. Apply slight tension to the lift yoke and visually verify proper disengagement of the lift yoke from the trunnions.

k. Remove the lift yoke, MPC lid and drain line from the pool in accordance with directions from the site's Radiation Protection personnel. Spray the equipment with demineralized water as they are removed from the pool.

Warning:
The MPC lid and unloaded MPC may contain residual contamination. All work done on the unloaded MPC should be carefully monitored and performed.

l. Disconnect the drain line from the MPC lid.

m. Store the MPC lid components in an approved location. Disengage the lift yoke from MPC lid. Remove any upper fuel spacers using the same process as was used in the installation.

n. Disconnect the Lid Retention System if used.

8.3.3 MPC Unloading

1. Remove the spent fuel assemblies from the MPC using applicable site procedures.

2. Vacuum the cells of the MPC to remove any debris or corrosion products.

3. Inspect the open cells for presence of any remaining items. Remove them as appropriate.

8.3.4 Post-Unloading Operations

1. Remove the HI-STAR 100 overpack and the unloaded MPC from the spent fuel pool as follows:

    a. Engage the lift yoke to the top trunnions.

    b. Apply slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the trunnions.

    c. Raise the HI-STAR 100 overpack until the HI-STAR 100 overpack flange is at the surface of the spent fuel pool.

ALARA Warning:
Activated debris may have settled on the top face of the HI-STAR 100 overpack during fuel unloading.

    d. Measure the dose rates at the top of the HI-STAR 100 overpack in accordance with plant radiological procedures and flush or wash the top surfaces to remove any highly-radioactive particles.

    e. Raise the top of the HI-STAR 100 overpack and MPC to the level of the spent fuel pool deck.

    f. Close the Annulus Overpressure System reservoir valve if the Annulus Overpressure System was used.

    g. Using a water pump, lower the water level in the MPC approximately 12 inches (30.5 cm) to prevent splashing during cask movement.

ALARA Note:
To reduce contamination of the HI-STAR 100 overpack, the surfaces of the HI-STAR 100 overpack and lift yoke should be kept wet until decontamination can begin.

    h. Remove the HI-STAR 100 overpack from the spent fuel pool while spraying the surfaces with plant demineralized water.

    i. Disconnect the Annulus Overpressure System from the HI-STAR 100 overpack via the quick disconnect. Drain the Annulus Overpressure System lines and reservoir.

    j. Place the HI-STAR 100 overpack in the designated preparation area.

k. Disengage the lift yoke.

l. Perform decontamination on the HI-STAR 100 overpack and the lift yoke.

2. Carefully decontaminate the area above the inflatable seal. Deflate, remove, and store the seal in an approved plant storage location.

3. Using a water pump, pump the remaining water in the MPC to the spent fuel pool or liquid radwaste system.

4. Drain the water in the annulus.

5. Remove the MPC from the HI-STAR 100 overpack and decontaminate the MPC as necessary.

6. Decontaminate the HI-STAR 100 overpack.

7. Remove any bolt plugs, seal surface protector and/or waterproof tape from the HI-STAR 100 overpack top bolt holes.

8. Move the HI-STAR 100 overpack and MPC for further inspection, corrective actions, or disposal as necessary.

LOCATION: ISFSI
RECOVER HI-STAR FROM STORAGE
LOCATION: CASK RECEIVING AREA
PLACE HI-STAR IN DESIGNATED PREPARATION AREA
LOCATION: CASK PREPARATION AREA
REMOVE HI-STAR CLOSURE PLATE
FILL ANNULUS
INSTALL ANNULUS SHIELD
REMOVE MPC CLOSURE RING
REMOVE MPC VENT PORT COVER PLATE AND SAMPLE MPC GAS
REMOVE MPC DRAIN PORT COVER PLATE
PERFORM MPC COOL-DOWN
FILL MPC CAVITY WITH WATER
REMOVE MPC LID TO SHELL WELD
INSTALL INFLATABLE SEAL
PLACE HI-STAR IN SPENT FUEL POOL
LOCATION: SPENT FUEL POOL
REMOVE MPC LID
DISCONNECT DRAIN LINE
REMOVE SPENT FUEL ASSEMBLIES FROM MPC
VACUUM CELLS OF MPC
REMOVE HI-STAR FROM SPENT FUEL POOL
LOCATION: CASK PREPARATION AREA
LOWER WATER LEVEL IN MPC
PUMP REMAINING WATER IN MPC TO SPENT FUEL POOL
REMOVE MPC FROM HI-STAR
DECONTAMINATE MPC AND HI-STAR

Figure 8.3.1; Unloading Operations Flow Diagram

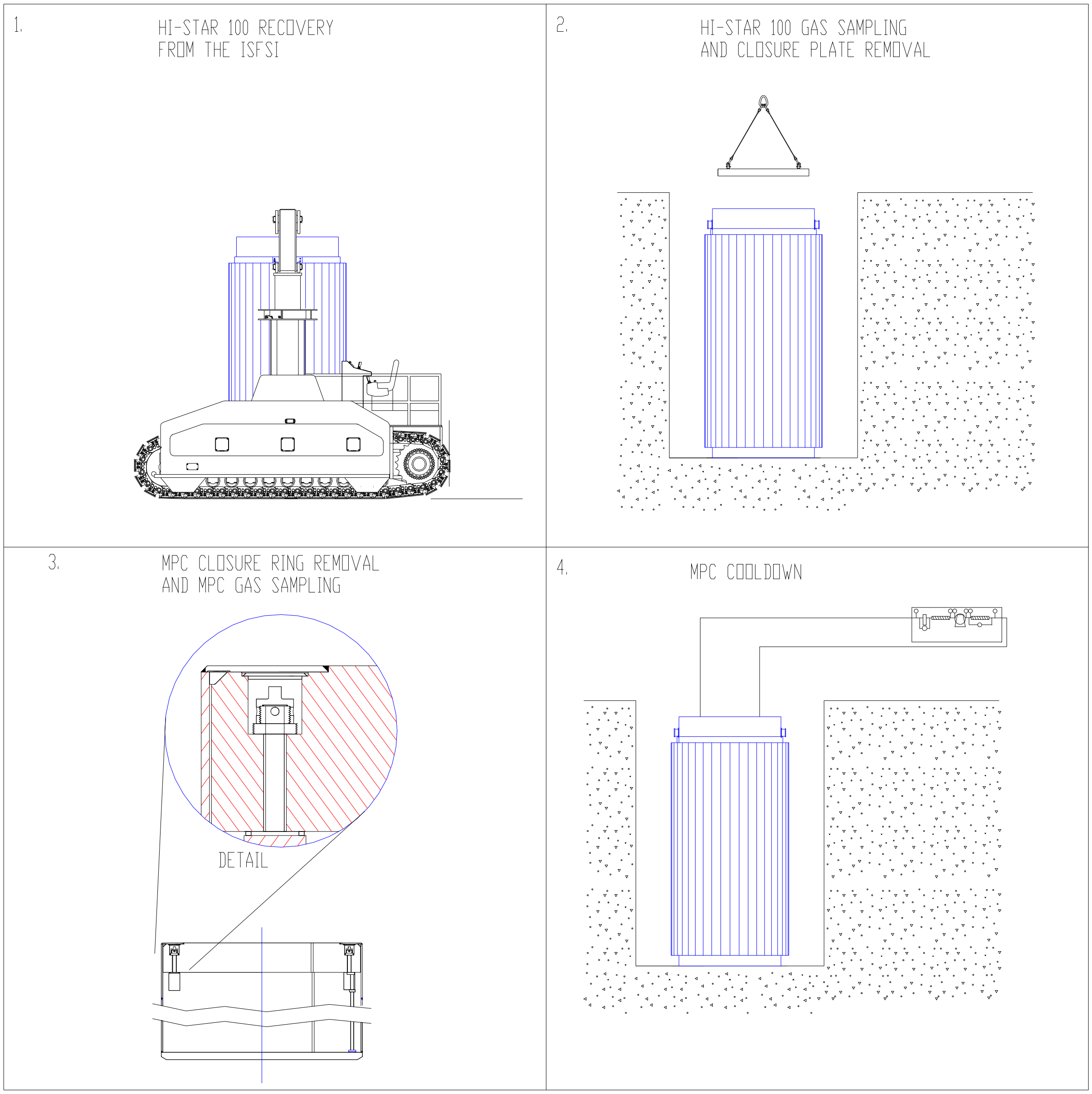

Figure 8.3.2a; Major HI-STAR 100 Unloading Operations (Sheet 1 of 3)

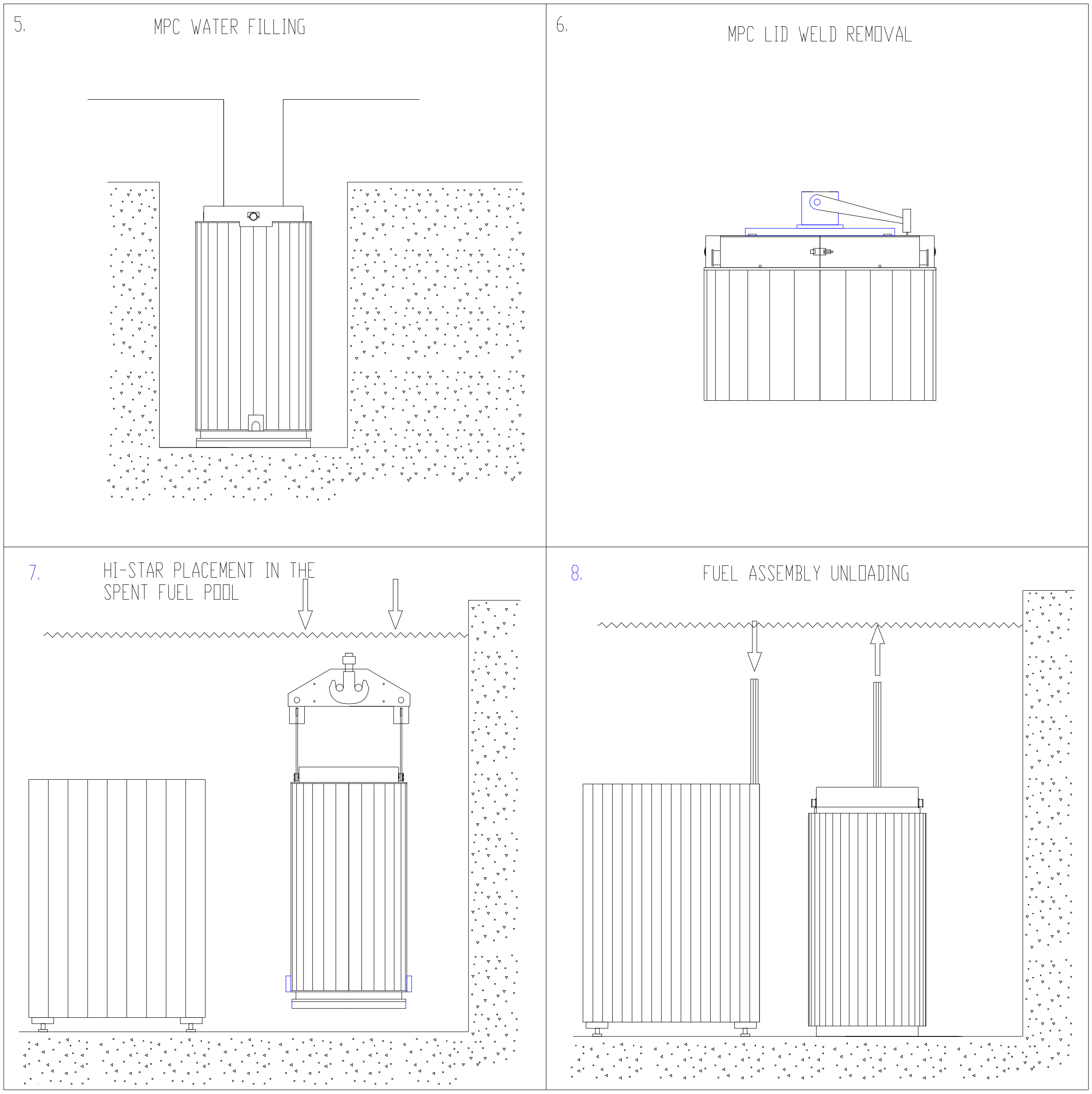

Figure 8.3.2b; Major HI-STAR 100 Unloading Operations (Sheet 2 of 3)

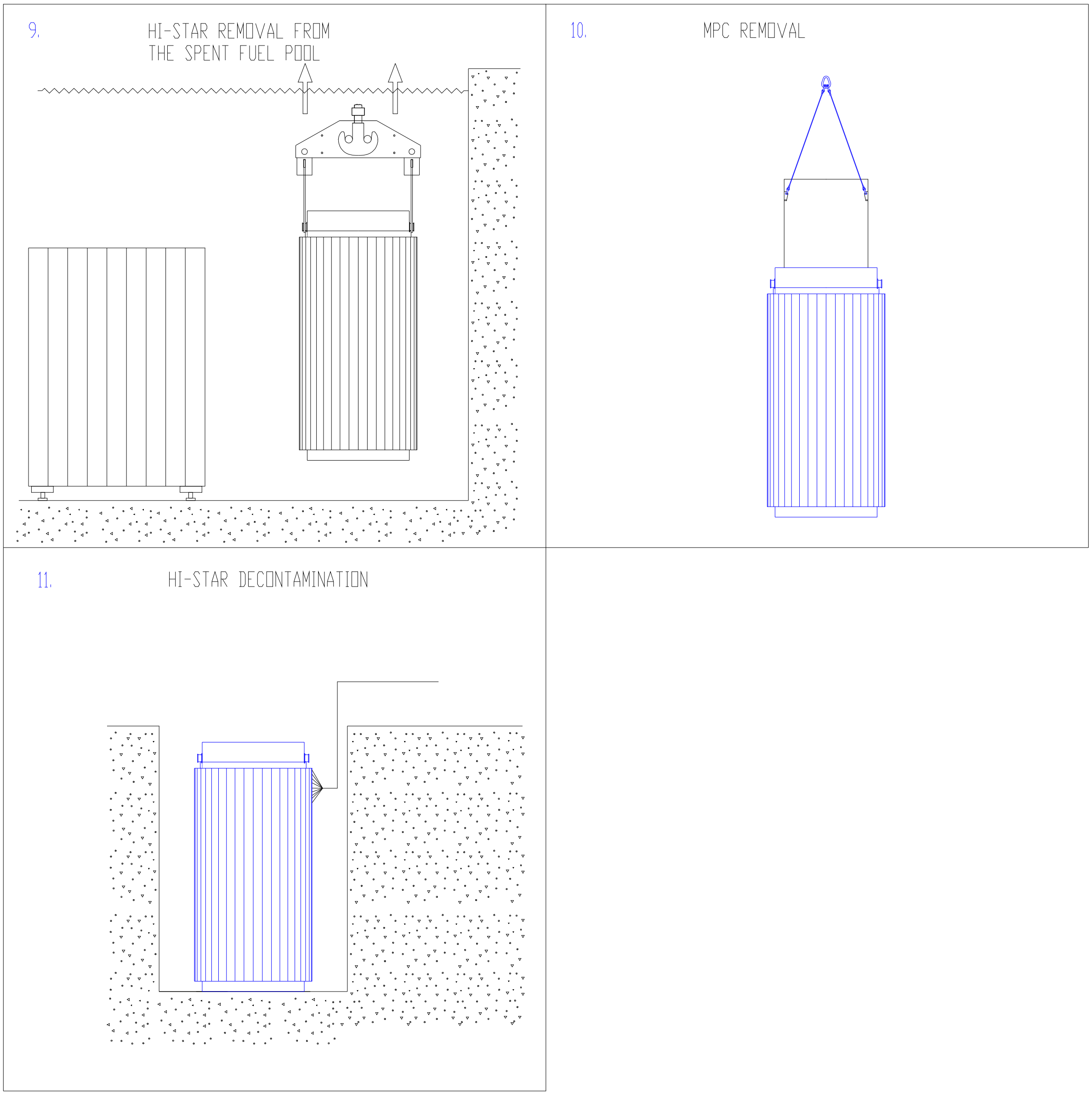

Figure 8.3.2c; Major HI-STAR 100 Unloading Operations (Sheet 3 of 3)

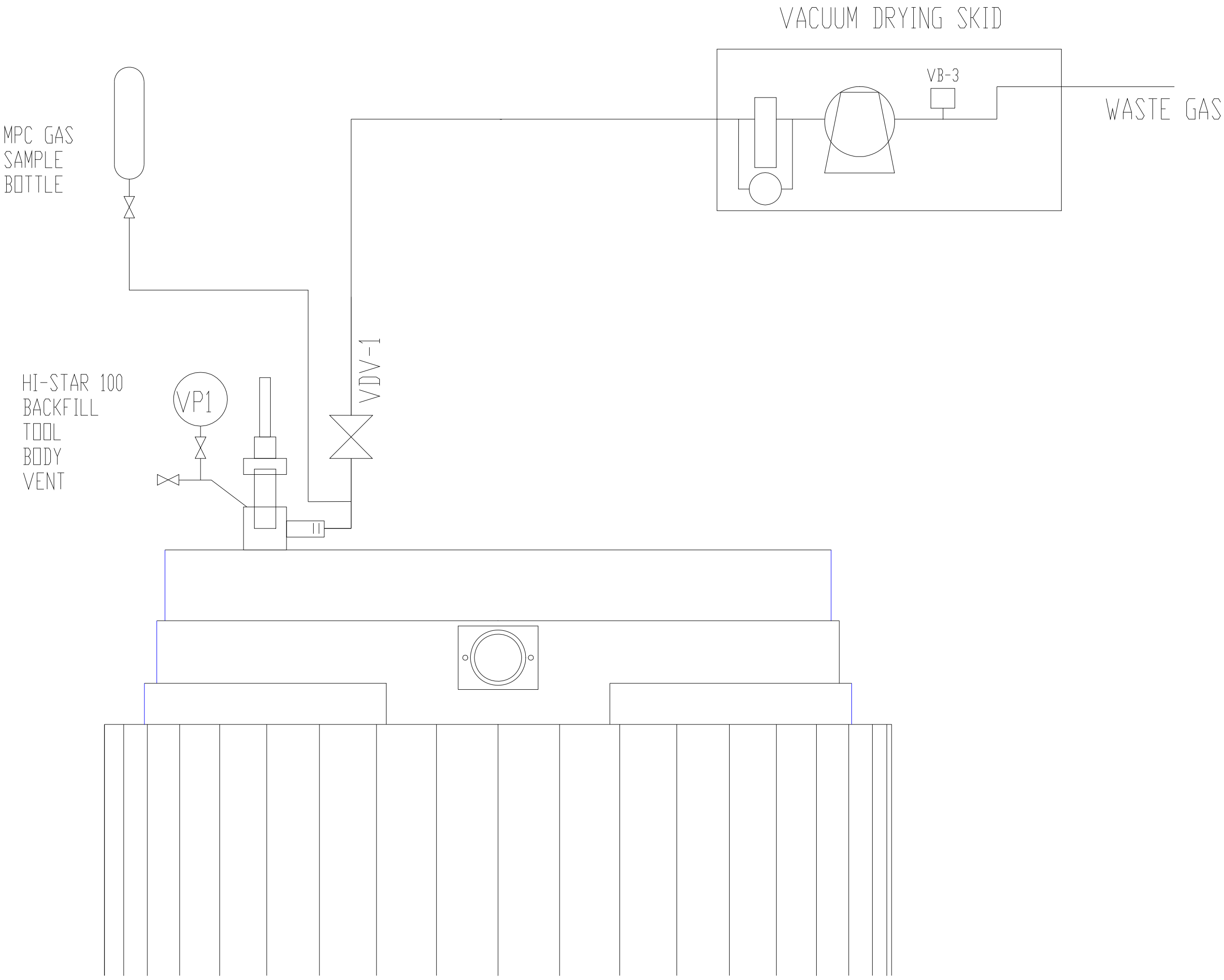

Figure 8.3.3; HI-STAR Annulus Gas Sampling

Figures 8.3.4 and 8.3.5 ; INTENTIONALLY DELETED

8.4 PLACEMENT OF THE HI-STAR 100 SYSTEM INTO STORAGE DIRECTLY FROM TRANSPORT

8.4.1 Overview of the HI-STAR 100 System Placement Operations Directly From Transport

The placement operations following transport of the overpack are similar to the later part of the loading operations detailed in Section 8.1. The overpack is received and surveyed for dose rates in accordance with 10CFR20 [8.4.1] and 10CFR49.173 to 177 [8.4.2]. The overpack is surveyed for removable contamination. The overpack may be transferred horizontally or vertically depending on the site specific requirements.

8.4.2 Storage Operations from Transport

1. Survey the overpack for dose rates to establish compliance with the Technical Specification limits.

2. Survey the overpack for removable contamination and decontaminate as necessary (to establish compliance with Technical Specification limits).

3. Transfer the overpack to the ISFSI.

Note:
The HI-STAR 100 minimum pitch shall be 12 feet (3.66 meters) (nominal).

4. Place the overpack at the approved storage location at the appropriate pitch.

5. Continue operations in accordance with Section 8.2.

8.5 REGULATORY ASSESSMENT

- The HI-STAR 100 System is compatible with wet and dry loading and unloading. General procedure descriptions for these operations are summarized in Sections 8.1 and 8.3 of the Topical Safety Analysis Report. Detailed procedures will need to be developed and evaluated on a site-specific basis.

- The bolted closure plate and welded MPC of the HI-STAR 100 System allow retrieval of the spent fuel for further processing or disposal as required.

- The smooth surfaces of the HI-STAR 100 System and its ancillary equipment are designed to facilitate decontamination. Only routine decontamination will be necessary after the HI-STAR 100 overpack is removed from the spent fuel pool.

- No significant radioactive effluents are produced during storage. Any radioactive effluents generated during the cask loading will be governed by the 10CFR Part 50 license conditions, if applicable.

- The general operating procedures described in the FSAR are adequate to protect health and minimize danger to life and property. Detailed procedures will need to be developed and evaluated on a site-specific basis.

- The operating procedures in the FSAR provide reasonable assurance that the HI-STAR 100 System will enable safe storage of spent fuel.

8.6 REFERENCES

[8.0.1] U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems", NUREG-1536, Final Report, January 1997.

[8.0.2] *U.S. Code of Federal Regulations,* Title 10, “Energy”, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

[8.0.3] American National Standard for Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 KG) or More for Nuclear Materials, ANSI N14.6, 1993.

[8.0.4] U.S. Nuclear Regulatory Commission, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612.

[8.0.5] Holtec International, “Final Safety Analysis Report for the HI-STORM 100 System”, Report HI-2002444, latest revision.

[8.0.6] Holtec International, “Safety Analysis Report on HI-STAR 100 Cask System”, Report HI-951251, latest revision.

[8.1.2] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment," ANSI N14.5-1997.

[8.1.3] American Society of Mechanical Engineers, “Boiler and Pressure Vessel Code”.

[8.2.1] Holtec International, "Topical Safety Analysis Report for the HI-STORM 100 System", Report HI-951312, latest revision.

[8.4.1] *U.S. Code of Federal Regulations,* Title 10, “Energy”, Part 20, “Standards for Protection Against Radiation.”

[8.4.2] *U.S. Code of Federal Regulations,* Title 49, “Transportation”, Part 173, “Shippers – General Requirements for Shipments and Packages.”

CHAPTER 9: ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM†

This chapter identifies the fabrication, inspection, test, and maintenance programs to be conducted on the HI-STAR 100 System *(overpack and MPC)* to verify that the structures, systems and components (SSCs) classified as important to safety have been fabricated, assembled, inspected, tested, accepted, and maintained in accordance with the requirements set forth in this FSAR, the applicable regulatory requirements, and the Certificate of Compliance. *The acceptance criteria and maintenance program requirements specified in this chapter apply to each HI-STAR 100 System fabricated, assembled, inspected, tested and accepted for use under the purview of the HI-STAR 100 System CoC.*

The controls, inspections, and tests set forth in this chapter, in conjunction with the design requirements described in previous chapters ensure that the HI-STAR 100 System will maintain confinement of radioactive material under normal, off-normal, and credible accident conditions; will maintain subcriticality control; *will reject the decay heat of the stored radioactive materials to the environment by passive means and maintain radiation doses within regulatory limits.*

Both pre-operational and operational tests and inspections are performed throughout HI-STAR 100 System operations to assure that the HI-STAR 100 System is functioning within its design parameters. These include receipt inspections, *nondestructive weld examinations, pressure test*, radiation shielding tests, thermal performance tests, dryness tests, and others. Chapter 8 identifies the sequence and conduct of the tests and inspections. "Pre-operation", as referred to in this section, defines that period of time from receipt inspection of a HI-STAR 100 System until the empty MPC is loaded into a HI-STAR overpack for fuel assembly loading.

The HI-STAR 100 System is classified as important-to-safety. Therefore, the individual structures, systems, and components (SSCs) that make up the HI-STAR 100 System shall be designed, fabricated, assembled, inspected, tested, accepted, and maintained in accordance with a quality program commensurate with the particular SSC's graded quality category. *The licensing drawings identify all important to safety subcomponents of the HI-STAR 100 System.*

The acceptance criteria and maintenance program described in this chapter fully comply with the requirements of 10CFR Part 72 [9.0.1] and NUREG-1536 [9.0.2] *to the maximum extent possible, as described in Chapter 1.*

The acceptance test requirements on the manufactured welds in the HI-STAR 100 System are contained in the component licensing drawing in Section 1.5. Additional details on the requirements in the drawing are provided in this chapter, which will be incorporated in the shop manufacturing documents (viz., weld procedures, shop travelers, inspection procedures, and fabrication procedures) to ensure full compliance with this FSAR.

Much of the new information in this chapter is directly extracted from previously NRC approved Holtec dockets; this information is shown in *italics*. In Chapter 9, this information was extracted from References [1.2.5 and 1.2.7]. All changes in this revision are marked with revision bars.

† Wherever multiple units are shown, the US units are the governing value, and the SI units are shown for information only.

9.1 ACCEPTANCE CRITERIA

This section provides the workmanship inspections and acceptance tests to be performed on the HI-STAR 100 System prior to or during first loading of the system. These inspections and tests provide assurance that the HI-STAR 100 System has been fabricated, assembled, inspected, tested, and accepted for use under the conditions specified in this FSAR and the Certificate of Compliance issued by the NRC in accordance with the requirements of 10CFR Part 72 [9.0.1].

These inspections and tests are also intended to demonstrate that the initial operation of the HI-STAR 100 System complies with the applicable regulatory requirements and the Technical Specifications. Noncompliances encountered during the required inspections and tests will be corrected or dispositioned to bring the item into compliance with this FSAR. Identification and resolution of noncompliances will be performed in accordance with the Holtec International Quality Assurance Program as described in Chapter 13 of this FSAR, or the licensee's NRC-approved Quality Assurance Program. The testing and inspection acceptance criteria applicable to the MPCs and the HI-STAR overpack are listed in Tables 9.1.1 and 9.1.2, respectively, and discussed in more detail in the sections that follow, and in Chapters 8 and 12. These inspections and tests are intended to demonstrate that the HI-STAR 100 System has been fabricated, assembled, and examined in accordance with the design criteria contained in Chapter 2 of this FSAR.

The contents of this chapter related to welding non-destructive examination are presented in the drawing package in Section 1.5. Likewise, the material on testing and maintenance of system components in this FSAR governs the content of the daughter documents such as the Manufacturing Manual and O&M Manual for the system components used in the manufacturing and long-term maintenance of the system components, respectively.

9.1.1 Fabrication and Nondestructive Examination (NDE)

This subsection summarizes the test program required for the HI-STAR 100 System.

9.1.1.1 Fabrication Requirements

The following fabrication controls and required inspections shall be performed on the HI-STAR 100 System, including the MPCs, in order to assure compliance to this FSAR and the Certificate of Compliance.

1. Materials of construction specified for the HI-STAR 100 System are identified in the drawing in Chapter 1 and shall be procured with certification and supporting documentation as required by ASME Code [9.1.1] Section II (where applicable); the requirements of ASME Code Section III (where applicable); Holtec procurement specifications; and 10CFR72, Subpart G. All materials and components will be receipt inspected for visual and dimensional acceptability, material conformance to specification requirements, and traceability markings, as applicable. Controls shall be in place to assure material traceability is maintained throughout fabrication.

Materials for the Confinement Boundary (MPC baseplate, lid, closure ring, port cover plates and shell) and the HI-STAR 100 System helium retention boundary (bottom plate, inner shell, top flange, vent and drain port plugs, closure plate, and closure plate bolts) (equivalent to the HI-STAR 100 System containment boundary in 10CFR71 [9.1.2] transport operations) shall also be procured per the requirements of ASME Section III, Article NB-2500.

2. The MPC Confinement Boundary and HI-STAR 100 System helium retention boundary shall be fabricated and inspected in accordance with ASME Code Section III, Subsection NB to the maximum extent practicable (see exceptions in Chapter 2). Other portions of the HI-STAR 100 overpack shall be fabricated and inspected in accordance with ASME Code Section III, Subsection NF (see exceptions in Chapter 2). The MPC basket, basket supports, and fuel spacers shall be fabricated and inspected in accordance with ASME Code Section III, Subsection NG (see exceptions in Chapter 2).

3. Welding shall be performed using welders and weld procedures that have been qualified in accordance with ASME Code Section IX and the applicable ASME Section III Subsections (e.g., NB, NG, or NF, as applicable to the SSC).

4. *Welds shall be examined as described in Section 9.1.1.4.*

5. The MPC confinement boundary and the HI-STAR overpack helium retention boundary shall be examined and tested by a combination of methods (including helium leak test, pressure test, *RT*, UT, MT and/or PT, as applicable) to verify that it is free of cracks, pinholes, uncontrolled voids or other defects that could significantly reduce its confinement effectiveness.

6. *Welds shall be repaired in accordance with Section 9.1.1.4.*

7. Any base metal repairs shall be performed and examined in accordance with the applicable fabrication Code.

8. Grinding and machining operations of the MPC Confinement Boundary and HI-STAR 100 helium retention boundary shall be controlled through written and approved procedures and quality assurance oversight to ensure grinding and machining operations do not reduce base metal wall thicknesses of the confinement or helium retention boundaries *below allowable limits*. The thicknesses of base metals shall be ultrasonically tested, as necessary, in accordance with written and approved procedures to verify base metal thickness meets *applicable* requirements. A nonconformance shall be written for areas found to be below allowable base metal thickness and shall be evaluated and repaired as necessary per the ASME Code Section III, Subsection NB requirements.

9. *Non-structural tack welds that do not become an integral part of a weld are not required to be removed. Non-structural tack welds that do not become integral part*

of a permanent weld shall be examined by an approved visual examination procedure.

10. Dimensional inspections of the HI-STAR 100 System shall be performed in accordance with written and approved procedures in order to verify compliance to design drawings and fit up of individual components. All dimensional inspections and functional fit-up tests shall be documented.

11. All required inspections shall be documented. The inspection documentation shall become part of the final quality documentation package.

12. The HI-STAR 100 System shall be inspected for cleanliness and proper packaging for shipping in accordance with written and approved procedures.

13. Each HI-STAR overpack will be durably marked with the appropriate model number, a unique identification number, and its empty weight per 10CFR72.236(k) at the completion of the acceptance test program.

14. A completed documentation package shall be prepared and maintained during fabrication of each HI-STAR 100 System to include detailed records and evidence that the required inspections and tests have been performed. *The completed document package shall* be reviewed to verify that the HI-STAR 100 System or component has been properly fabricated and inspected in accordance with the design and Code construction requirements. The documentation package shall include, but not be limited to:

    - Completed Weld Records
    - Inspection Records
    - Nonconformance Reports
    - Material Test Reports
    - NDE Reports
    - Dimensional Inspection Reports

9.1.1.2 <u>MPC Lid-to-Shell Weld Inspection</u>

1. The MPC lid-to-shell (LTS) weld shall be multi-layer liquid penetrant (PT) examined following completion of welding.

2. Deleted.

3. A multi-layer PT examination shall be performed. The multi-layer PT must, at a minimum, include the root and final layers and one intermediate PT after each approximately 3/8 inch (9.5 mm) of weld depth has been completed. The 3/8 inch (9.5 mm) weld depth corresponds to the maximum allowable flaw size in the weld [9.1.10].

4. The overall minimum thickness of the LTS weld has been increased by 0.125 inch (3.2 mm) over the size credited in structural analyses, to provide additional structural capacity. A 0.625-inch (15.9 mm) J-groove weld was assumed in the structural analyses in Chapter 3.

5. For PT, the maximum undetectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME Section XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded. The inspection process, including findings (indications) shall be made a permanent part of the cask user's records by video, photographic, or other means which provide an equivalent retrievable record of weld integrity. *Mapping is considered an equivalent record which contains the type, size and location of the relevant indications discovered during weld examination. The documentation of relevant indications should be taken during the final interpretation period described in ASME Section V, Article 6, T-676*. The inspection of the weld shall be performed by qualified personnel and shall meet the acceptance requirements of ASME Section III, NB-5350 for PT.

6. Evaluation of any indications identified by non-destructive examination shall include consideration of any active flaw mechanisms. However, cyclic loading *on the LTS weld* is not significant, so fatigue will not be a factor. The LTS weld is protected from the external environment by the closure ring and the root of the LTS weld is dry and inert (He atmosphere), so stress corrosion cracking will not be a concern for the LTS weld.

7. The multi-layer PT examination of the LTS weld, in conjunction with *the discussion in Chapter 7*, provide reasonable assurance that the LTS weld is sound and will perform its design function under all loading conditions.

9.1.1.3 *Visual Inspections and Measurements*

1. The HI-STAR 100 System components shall be assembled in accordance with the licensing drawing package in Section 1.5. The drawings provide dimensional tolerances that define the limits on the dimensions used in licensing basis analysis. Fabrication drawings provide additional dimensional tolerances necessary to ensure fit-up of parts. Visual inspections and measurements shall be made and controls shall be exercised to ensure that the cask components conform to the dimensions and tolerances specified on the licensing and fabrication drawings. These dimensions are subject to independent confirmation and documentation in accordance with the Holtec QA program.

The following shall be verified as part of visual inspections and measurements:

- *Visual inspections and measurements shall be made to ensure that the systems' effectiveness is not significantly reduced as a result of manufacturing deviations. Any important-to-safety component found to be under the specified minimum thickness shall be justified under the rules of 10CFR72.48 or repaired or replaced, as appropriate.*
- *Visual inspections shall be made to verify that neutron absorber panels and basket shims are present as required by the MPC basket design.*
- *The system components shall be inspected for cleanliness and preparation for shipping in accordance with written and approved procedures.*

The visual inspection and measurement results for the HI-STAR 100 System shall become part of the final quality documentation package.

9.1.1.4 Weld Examination

The examination of the HI-STAR 100 System welds shall be performed in accordance with the drawing package in Section 1.5 and the applicable codes and standards.

All code weld inspections shall be performed in accordance with written and approved procedures by personnel qualified in accordance with SNT-TC-1A. All required inspections, examinations, and tests shall become part of the final quality documentation package.

The following specific weld requirements shall be followed in order to verify fabrication in accordance with the provisions of this FSAR.

1. *Confinement Boundary welds including any attachment welds (and temporary welds to the Confinement Boundary) shall be examined in accordance with ASME Code Section V, with acceptance criteria per ASME Code Section III, Subsection NB, Article NB-5300. Examinations, Visual (VT), Radiographic (RT), and Liquid Penetrant (PT), apply to these welds as defined by the code. These welds shall be repaired in accordance with the requirements of the ASME Code Section III, Article NB-4450 and examined after repair in the same manner as the original weld.*

2. *Basket welds shall be VT examined and repaired in accordance with written and approved procedures developed for welding Metamic and in accordance with the philosophy of ASME Section IX.*

3. *Non-code welds shall be examined and repaired in accordance with written and approved procedures as defined in the system Manufacturing Manual.*

9.1.2 Structural and Pressure Tests

9.1.2.1 Lifting Trunnions

Two trunnions (located near the top of the HI-STAR overpack) are provided for vertical lifting and handling of the HI-STAR 100 System. The trunnions are designed in accordance with ANSI N14.6 [9.1.5] using a high-strength and high-ductility material (see Chapter 1). The trunnion contains no welded components. The maximum design lifting load of 250,000 pounds (113,398 kg) for the HI-STAR 100 System will occur during the removal of the HI-STAR overpack from the spent fuel pool after the MPC has been loaded, flooded with water, and the MPC lid is installed. The high material ductility, absence of materials vulnerable to brittle fracture, large stress margins, and a carefully engineered design to eliminate local stress raisers in the highly stressed regions (during the lift operation) ensure that the lifting trunnions will work reliably. However, pursuant to the defense-in-depth approach of NUREG-0612 [9.1.6], the acceptance criteria for the lifting trunnions must be established in conjunction with other considerations applicable to heavy load handling.

Section 5 of NUREG-0612 calls for measures to "provide an adequate defense-in-depth for handling of heavy loads...". The NUREG-0612 guidelines cite four major causes of load handling accidents, of which rigging failure (including trunnion failure) is one:

i. operator errors
ii. rigging failure
iii. lack of adequate inspection
iv. inadequate procedures

The cask loading and handling operations program shall ensure maximum emphasis to mitigate the potential of load drop accidents by implementing measures to eliminate shortcomings in all aspects of the operation including the four aforementioned areas.

In order to ensure that the lifting trunnions do not have any hidden material flaws, the trunnions shall be tested at 300% of the maximum design (service) lifting load. The load (750,000 lbs (340,194 kg)) shall be applied for a minimum of 10 minutes. The accessible parts of the trunnions (areas outside the HI-STAR overpack), and the local HI-STAR 100 cask areas will then be visually examined to verify no deformation, distortion, or cracking has occurred. Any evidence of deformation, distortion or cracking of the trunnion or adjacent HI-STAR 100 cask areas will require replacement of the trunnion and/or repair of the HI-STAR 100 cask. Following any replacements and/or repair, the load testing shall be reperformed and the components re-examined in accordance with the original procedure and acceptance criteria. Testing will be performed in accordance with written and approved procedures. Certified material test reports verifying trunnion material mechanical properties meet ASME Code Section II requirements will provide further verification of the trunnion load capabilities. Test results shall be documented. The documentation shall become part of the final quality documentation package.

The acceptance testing of the trunnions in the manner described above will provide adequate assurance against handling accidents.

9.1.2.2 Pressure Testing

9.1.2.2.1 HI-STAR 100 Helium Retention Boundary

The helium retention boundary of the HI-STAR overpack (e.g., the containment boundary during transportation) will be hydrostatically or pneumatically pressure tested . *The test requirements for pressure testing are shown in Table 2.0.2*. The test shall be performed in accordance with written and approved procedures. The approved test procedure shall clearly define the test equipment arrangement.

The overpack pressure test may be performed at any time during fabrication after the containment boundary is complete. Preferably, the pressure test should be performed after all overpack fabrication is complete, including attachment of the intermediate shells. The HI-STAR overpack shall be assembled for this test with the closure plate mechanical seal (only one required) or temporary test seal installed. Closure bolts shall be installed and torqued to an appropriate value less than or equal to the value specified in Chapter 8.

The calibrated test pressure gage installed on the overpack shall have an upper limit of approximately twice that of the test pressure. The test pressure shall be maintained for ten minutes. During this time period, the pressure gage shall not fall below *test pressure*. At the end of ten minutes, and while the pressure is being maintained at a minimum of *test pressure*, the overpack shall be observed for leakage. In particular, the closure plate-to-top forging joint (the only credible leakage point) shall be examined. If a leak is discovered, the overpack will be emptied and an evaluation to determine the cause of the leakage will be made. Repairs and retest shall be performed until the pressure test criteria are met.

> Note: If failure of the pressure retest occurs after initial repairs are completed, a nonconformance report shall be issued and root cause and corrective action shall be addressed before further repairs and retest are performed.

After completion of the pressure testing, the closure plate will be removed and the internal surfaces shall be visually examined for cracking or deformation. Any evidence of cracking or deformation shall be cause for rejection or repair and retest, as applicable. The overpack shall be required to be pressure tested until all examinations are found to be acceptable. All test results shall be documented and shall become part of the final quality documentation package.

9.1.2.2.2 <u>MPC Confinement Boundary</u>

Pressure testing (hydrostatic or pneumatic) of the MPC Confinement Boundary shall be performed to verify the lid-to-shell field weld in accordance with the requirements of the ASME Code Section III, Subsection NB, Article NB-6000 and applicable sub-articles, when field welding of the MPC lid-to-shell weld is completed. The test requirements for pressure testing are shown in Table 2.0.1. The calibrated test pressure gage installed on the MPC Confinement Boundary shall have an upper limit of approximately twice that of the test pressure. The MPC vent and drain ports will be used for pressurizing the MPC cavity. Water shall be pumped into the MPC drain port until water only is flowing from the MPC vent port. The MPC vent port is then closed and the pressure is increased to the test pressure. While the MPC is under pressure, the MPC lid-to-shell weld shall be examined for leakage. If any leaks are observed, the pressure shall be released and the weld shall be repaired in accordance with the requirements of ASME Code, Section III, Subsection NB. Following completion of the required hold period at the test pressure, the pressure shall be released and the surface of the MPC lid-to-shell weld shall be re-examined by liquid penetrant examination in accordance with

ASME Code, Section III, Subsection NB, Article NB-5350 acceptance criteria. Any evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable.
If a leak is discovered, the test pressure shall be reduced, the MPC cavity water level lowered, if applicable, the MPC cavity vented, and the weld shall be examined to determine the cause of the leakage and/or cracking. Repairs to the weld shall be performed in accordance with approved written procedures prepared in accordance with the ASME Code Section III, Subsection NB, NB-4450.

The MPC Confinement Boundary pressure test shall be repeated until all required examinations are found to be acceptable in accordance with the acceptance criteria. All test results shall be documented and shall be maintained as part of the loaded MPC quality documentation package.

9.1.2.3 Materials Testing

The majority of material used in the HI-STAR overpack are ferritic steels. ASME Code Section II *and Section III require that certain materials be tested in order to assure that these materials are not subject to brittle fracture failures.*

Each plate or forging for the helium retention boundary (overpack inner shell, bottom plate, top flange, and closure plate) shall be required to be drop weight tested in accordance with the requirements of Regulatory Guides 7.11 and 7.12, as applicable. Additionally, per the ASME Code Section III, Subsection NB, Article NB-2300, Charpy V-notch testing shall be performed on these materials. Weld material used in welding the helium retention boundary shall be Charpy V-notch tested in accordance with ASME Section III, Subsection NB, Articles NB-2300 and NB-2430.

Non-helium retention boundary portions of the overpack, as required, shall be Charpy V-notch tested in accordance with ASME Section III, Subsection NF, Articles NF-2300, and NF-2430. The non-helium retention boundary materials to be tested include the intermediate shells, overpack port cover plates, and applicable weld materials. Section 3.1 provides the test temperatures or T_{NDT}, and test requirements to be used when performing the testing specified above.

All test results shall be documented and shall become part of the final quality documentation package.

9.1.2.4 Pneumatic Testing of the Neutron Shield Enclosure Vessel

A pneumatic pressure test of the neutron shield enclosure vessel will be performed following final closure welding of the enclosure shell returns and enclosure panels. The pneumatic test pressure *requirements are shown in Table 9.1.5*. The test shall be performed in accordance with approved written procedures.

During the test, the two rupture disks on the neutron shield enclosure vessel will be removed. One of the rupture disk threaded connections will be used for connection of the air pressure line and the other rupture disk connection will be used for connection of the pressure gauge.

Following introduction of pressurized air into the neutron shield enclosure vessel, a 10 minute pressure hold time will be required. If the neutron shield enclosure vessel fails to hold pressure, an approved soap bubble solution will be applied to determine the location of the leak. The leak shall be repaired using weld repair procedures in accordance with the ASME Code Section III, Subsection NF, Article NF-4450. The pneumatic pressure test shall be re-performed until no pressure loss is observed.

All test results shall be documented and shall become part of the final quality documentation package.

9.1.3 Leakage Testing

Leakage testing shall be performed in accordance with written and approved procedures and the leakage test methods and procedures of ANSI N14.5 [9.1.9], as follows.

Helium leakage testing of the MPC base metals (shell, baseplate, and MPC lid) and MPC shell to baseplate and shell to shell welds is performed on the unloaded MPC. The acceptance criterion is defined in *Table 2.0.1*. The helium leakage test of the vent and drain port cover plate welds shall be performed using a helium mass spectrometer leak detector (MSLD). If a leakage rate exceeding the acceptance criterion is detected, then the area of leakage shall be determined and the area repaired per ASME Code Section III, Subsection NB, Article NB-4450 requirements. Re-testing shall be performed until the leakage rate acceptance criterion is met.

Leakage testing of the field welded MPC lid-to-shell weld and closure ring welds are *as described in Chapter 7*. Leak testing results for the MPC shall be documented and shall become part of the quality record documentation package.

Leakage testing of the vent and drain port cover plate welds shall be performed after welding of the cover plates and subsequent NDE.

9.1.3.1 HI-STAR Overpack

A helium retention boundary weld leakage test shall be performed at any time after the containment boundary fabrication is complete. Preferably, this test should be performed at the completion of overpack fabrication, after all intermediate shells have been attached. The leakage test *requirements are shown in Table 2.0.2*. If a leakage rate exceeding the acceptance criteria is detected, the area of leakage shall be determined using the sniffer probe method or other means, and the area will be repaired per ASME Code Section III, Subsection NB, NB-4450 requirements. Following repair and appropriate NDE, the leakage testing shall be re-performed until the test criteria are satisfied.

Note: If failure of the leakage rate retest occurs after initial repairs are completed, a nonconformance report shall be issued and root cause and corrective action shall be addressed before further repairs and retest are performed.

At the completion of overpack fabrication, helium leakage through the helium retention penetrations (consisting of the inner mechanical seal between the closure plate and top flange and the vent and drain port plug seals) shall be demonstrated to not exceed the leakage rate and minimum test sensitivity *shown in Table 2.0.2*. This may be performed simultaneously with the boundary weld leakage test or may be performed separately using the methods described in the paragraph below.

Testing of the helium retention penetrations may be performed by evacuating and backfilling the overpack with helium gas. . A helium MSLD will be used (see Chapter 8 for details of test connections specifically designed for testing the penetration seals) to perform the test. Starting with the vent or drain port plug, the test cover is connected. The cavity on the external side of the vent port plug is evacuated and the vacuum pump is valved out. The MSLD detector measures the leakage rate of helium into the test cavity. The minimum test sensitivity shall be *per Table 2.0.2*. If the leakage rate exceeds the acceptance *criteria shown in Table 2.0.2,* the test chamber is vented and removed. The corresponding plug seal is removed, seal seating surfaces are inspected and cleaned, and the plug with a new seal is reinstalled and torqued to the required value. The test process is then repeated until the seal leakage rate is successfully achieved. The same process is repeated for the remaining overpack vent or drain port. The process is also used to test the closure plate seal except that the closure plate test tool (see Chapter 8 for details) is used in lieu of the test cover.

If the total measured leakage rate for all tested penetrations does not exceed *the acceptance criteria shown in Table 2.0.2*, the leakage tests are successful. If the total leakage rate exceeds *the acceptance criteria*, an evaluation should be performed to determine the cause of the leakage, repairs made as necessary, and the overpack must be re-tested until the total leakage rate is within the required acceptance criterion. All leak testing results for the HI-STAR overpack shall become part of the quality record documentation package.

9.1.4 Component Tests

9.1.4.1 Valves, Rupture Disks, and Fluid Transport Devices

There are no fluid transport devices associated with the HI-STAR 100 System. The only valve-like components in the HI-STAR 100 System are the specialty designed caps installed in the MPC lid for the drain and vent ports. These caps are recessed inside the MPC lid and covered by the fully welded vent and drain port cover plates. No credit is taken for the caps' ability to confine helium or radioactivity. After completion of drying and backfill operations, the drain and vent port cover plates are welded in place on the MPC lid and *are liquid penetrant examined and leakage tested* to verify the MPC confinement boundary.

There are two rupture disks installed in the upper ledge surface of the neutron shield enclosure vessel of the HI-STAR overpack. These rupture disks are provided for venting purposes under hypothetical fire accident conditions in which vapor formation from neutron shielding material degradation may occur. *The rupture disks relief data is shown in Table 9.1.5*. Each manufactured lot of rupture disks shall be sample tested to verify their point of rupture.

9.1.4.2 Seals and Gaskets

Two metallic mechanical seals are provided on the HI-STAR overpack closure plate to provide *redundant sealing. Mechanical seals* are also used on the overpack vent and drain port plugs of the HI-STAR overpack. *Each primary* seal is individually leak tested in accordance with Subsection 9.1.3. An independent and redundant seal is provided for each penetration (e.g., closure plate, port cover plates, and closure plate test plug). No confinement credit is taken for these redundant seals and they are not leakage tested. Details on these seals are provided in Chapter 7. Procedures for leakage testing are provided in Chapter 8.

9.1.5 Shielding Integrity

The HI-STAR 100 System has three specifically designed shields for neutron and gamma ray attenuation. For gamma shielding, there are successive carbon steel intermediate shells attached onto the outer surface of the overpack inner shell. The details of the manufacturing process are discussed in Chapter 1. Holtite-A neutron shielding is provided in the outer enclosure of the overpack. Additional neutron attenuation is provided by the neutron absorber attached to the fuel basket cell surfaces inside the MPCs. Test requirements for each of the three shielding items are described below.

9.1.5.1 Fabrication Testing and Controls

Holtite-A:

Neutron shield properties of Holtite-A are provided in Chapter 1. Each manufactured lot of neutron shield material shall be tested to verify that the material composition (aluminum and hydrogen), boron concentration, and neutron shield density (or specific gravity) meet the requirements specified in Chapter 1 and the Bill of Material sections. A manufactured lot is defined as the total amount of material used to make any number of mixed batches comprised of constituent ingredients from the same lot/batch identification numbers supplied by the constituent manufacturer. Testing will be performed in accordance with written and approved procedures and/or standards. Material composition, boron concentration, and density (or specific gravity) data for each manufactured lot of neutron shield material will become part of the quality documentation package.

The installation of the neutron shielding material shall be performed in accordance with written and qualified procedures. The procedures shall ensure that mix ratios and mixing methods are controlled in order to achieve proper material composition, boron concentration and distribution, and that pours are controlled in order to prevent gaps from occurring in the material. Samples of each manufactured lot of neutron shield material will be maintained by Holtec International as part of the quality record documentation package.

Steel:

All steel plates utilized in the construction of the HI-STAR 100 System shall be dimensionally inspected to assure compliance for minimum thickness in accordance with the Design Drawings in Section 1.5.

The total measured thickness of the inner shell plus intermediate shells shall be a minimum of 8.5 inches (215.9 mm). The top flange, closure plate, and bottom plate of the overpack shall be measured to confirm their thicknesses meet design drawing requirements. Measurements shall be performed in accordance with written and approved procedures. The measurement locations and measurements shall be documented. Measurements shall be made through a combination of receipt inspection thickness measurements on individual plates and actual measurements taken prior to welding the overpack or intermediate shells. Any area found to be under the specified minimum thickness will be repaired in accordance with applicable ASME Code requirements.

No additional gamma shield testing of the HI-STAR 100 System is required. A gamma shielding effectiveness test per Subsection 9.1.5.2 will be performed on each fabricated HI-STAR 100 System after the first fuel loading.

General for All Shield Materials:

1. All test results shall be documented and become part of the quality documentation package.

2. Dimensional inspections of the cavities containing poured neutron shielding materials shall assure that the design required amount of shielding material is incorporated into the fabricated item.

9.1.5.2 Shielding Effectiveness Test

Following the first fuel loading of each HI-STAR 100 System, a shielding effectiveness test will be performed at the loading facility site to verify the effectiveness of the gamma and neutron shields. *This test will be performed after the HI-STAR 100 System has been loaded with an MPC containing spent fuel assemblies and the MPC has been drained, moisture removed, and backfilled with helium*.

The neutron and gamma shielding effectiveness tests *shall* be performed *after fuel loading* using written and approved procedures. Calibrated neutron and gamma dose meters shall be used to measure the actual neutron and gamma dose rates at the surface of the HI-STAR overpack. Measurements *shall* be taken at *the locations specified in the Radiation Protection Program for comparison against the prescribed limits.* The test is considered acceptable if the dose *rate* readings are *less than* or equal to the *calculated limits.* If dose rates are higher than the limits, the required actions *provided in the Radiation Protection Program shall be completed. Dose rate measurements shall be documented and shall become part of the quality documentation package.*

9.1.5.3 Neutron Absorber Tests

Each plate of neutron absorber shall be visually inspected for damage such as scratches, cracks, burrs, peeled cladding, foreign material embedded in the surfaces, voids, delamination, and surface finish, as applicable. These testing requirements are in accordance with HI-STORM 100 FSAR.

9.1.5.3.1 Boral (75% Credit)

After manufacturing, a statistical sample of each lot of neutron absorber shall be tested using wet chemistry and/or neutron attenuation testing to verify a minimum ^{10}B content (areal density) in samples taken from the ends of the panel. The minimum ^{10}B loading of the neutron absorberl panels for each MPC model is provided in Table 2.1.17. Any panel in which ^{10}B loading is less than the *minimum allowed shall be rejected. Testing shall be performed using written and approved procedures. Results shall be documented and become part of the cask quality records documentation package.*

9.1.5.3.2 <u>METAMIC® (90% Credit)</u>

NUREG/CR-5661 identifies the main reason for a penalty in the neutron absorber B-10 density as the potential of neutron streaming due to non-uniformities in the neutron absorber, and recommends comprehensive acceptance tests to verify the presence and uniformity of the neutron absorber for credits more than 75%. Since a 90% credit is taken for METAMIC®, the following criteria must be satisfied:

- *The boron carbide powder used in the manufacturing of METAMIC® must have small particle sizes to preclude neutron streaming*
- *The ^{10}B areal density must comply with the limits of Table 2.1.17.*
- *The B_4C powder must be uniformly dispersed locally, i.e. must not show any particle agglomeration. This precludes neutron streaming.*
- *The B_4C powder must be uniformly dispersed macroscopically, i.e. must have a consistent concentration throughout the entire neutron absorber panel.*
- *The maximum B_4C content in METAMIC® shall be less than or equal to 33.0 weight percent.*

To ensure that the above requirements are met the following tests shall be performed:

- *All lots of boron carbide powder are analyzed to meet particle size distribution requirements.*
- *The following qualification testing shall be performed on the first production run of METAMIC® panels for the MPCs in order to validate the acceptability and consistency of the manufacturing process and verify the acceptability of the METAMIC® panels for neutron absorbing capabilities:*

    1) *The boron carbide powder weight percent shall be verified by testing a sample from forty different mixed batches. (A mixed batch is defined as a single mixture of aluminum powder and boron carbide powder used to make one or more billets. Each billet will produce several panels.) The samples shall be drawn from the mixing containers after mixing operations have been completed. Testing shall be performed using the wet chemistry method.*

    2) *The ^{10}B areal density shall be verified by testing a sample from one panel from each of forty different mixed batches. The samples shall be drawn from areas contiguous to the manufactured panels of METAMIC® and shall be tested using the wet chemistry method. Alternatively, or in addition to the wet chemistry tests, neutron attenuation tests on the samples may be performed to quantify the actual ^{10}B areal density.*

3) *To verify the local uniformity of the boron particle dispersal, neutron attenuation measurements of random test coupons shall be performed. These test coupons may come from the production run or from pre-production trial runs.*

4) *To verify the macroscopic uniformity of the boron particle distribution, test samples shall be taken from the sides of one panel from five different mixed batches before the panels are cut to their final sizes. The sample locations shall be chosen to be representative of the final product. Wet chemistry or neutron attenuation shall be performed on each of the samples.*

- *During production runs, testing of mixed batches shall be performed on a statistical basis to verify the correct boron carbide weight percent is being mixed.*

- *During production runs, samples from random METAMIC® panels taken from areas contiguous to the manufactured panels shall be tested via wet chemistry and/or neutron attenuation testing to verify the ^{10}B areal density. This test shall be performed to verify the continued acceptability of the manufacturing process.*

The measurements of B_4C particle size, ^{10}B isotopic assay, uniformity of B_4C distribution and ^{10}B areal density shall be made using written and approved procedures. Results shall be documented.

9.1.5.3.3 Installation of the Neutron Absorber Panels

Installation of neutron absorber panels into the fuel basket shall be performed in accordance with written and approved instructions. Travelers and quality control procedures shall be in place to assure each required cell wall of the MPC basket contains a neutron absorber panel in accordance with drawings in Chapter 1. These quality control processes, in conjunction with in-process manufacturing testing, provide the necessary assurances that the neutron absorber will perform its intended function. No additional testing or in-service monitoring of the neutron absorber material will be required.

9.1.6 Thermal Acceptance Test

The first fabricated HI-STAR overpack shall be tested to confirm its heat transfer capability. The test shall be conducted after the radial channels, enclosure shell panels, and neutron shield material have been installed and all inside and outside surfaces are painted per the design drawings. A test cover plate shall be used to seal the overpack cavity. Testing shall be performed in accordance with written and approved procedures.

The thermal test is performed by heating the overpack cavity with a readily measurable source of thermal energy. Prior standard practice has utilized electrical heating systems for confirming thermal performance of casks. However, as explained below, the HI-STAR 100 overpack thermal acceptance test is performed using steam as the source of thermal energy. Steam heating of the overpack cavity surfaces is the preferred method for this test instead of electric heating. There are several advantages with steam heated testing as listed below:

(i) Uniform cavity surface temperatures are readily achieved as a result of high steam condensation heat transfer coefficient (about 1,000 Btu/ft^2 hr-^{o}F (5678 W/m^2 K) compared to about 1 Btu/ft^2 hr-^{o}F (5.678 W/m^2 K) for air) coupled with the steam's uniform distribution throughout the cavity.

(ii) A reliable constant temperature source (steam at atmospheric pressure condenses at 212^o F (100^o C) compared to variable heater surface temperatures in excess of 1,000^o F (537.7^o C)) eliminates concerns of overpack cavity surface overheating.

(iii) Interpretation of isothermal test data is not susceptible to errors associated with electric heating systems due to heat input measurement uncertainties, leakage of heat from electrical cables, thermocouple wires, overpack lid, bottom baseplate, etc.

(iv) The test setup is simple requiring only a steam inlet source and drain compared to numerous power measurement and control instruments, switchgear and safety interlocks required to operate an electric heater assembly.

Twelve (12) calibrated thermocouples shall be installed on the external walls of the overpack as shown in Figure 9.1.2. Three calibrated thermocouples shall be installed on the internal walls of the overpack in locations to be determined by procedure. Additional temperature sensors shall be used to monitor ambient temperature, steam supply temperature, and condensate drain temperature. The thermocouples shall be attached to strip chart recorders or other similar mechanism to allow for continuous monitoring and recording of temperatures during the test. Instrumentation shall be installed to monitor overpack cavity internal pressure.

After the thermocouples have been installed, dry steam will be introduced through an opening in the test cover plate previously installed on the overpack and the test initiated. Temperatures of the thermocouples, plus ambient, steam supply, and condensate drain temperature shall be recorded at hourly intervals until thermal equilibrium is reached. Appropriate criteria defining when thermal equilibrium is achieved shall be determined based on a variety of potential ambient test conditions and incorporated into the test procedure. In general, thermal equilibrium is expected approximately 12 hours after the start of steam heating. Air will be purged from the overpack cavity via venting during the heatup cycle. During the test, the steam condensate flowing out of the overpack drain shall be collected and the mass of the condensate measured with a precision weighing instrument.

Once thermal equilibrium is established, the final ambient, steam supply, and condensate drain temperatures and temperatures at each of the thermocouples shall be recorded. The strip charts, hand-written logs, or other similar readout shall be marked to show the point when thermal equilibrium was established and final test measurements were recorded. The final test readings along with the hourly data inputs and strip charts (or other similar mechanism) shall become part of the quality records documentation package for the overpack. The heat rejection capability of the overpack at test conditions shall be computed using the following formula:

$$Q_{hm} = (h_1 \text{ B } h_2)\, m_c \qquad (8\text{-}1)$$

Where: Q_{hm} = Heat rejection rate of the overpack (Btu/hr)

h_1 = Enthalpy of steam entering the overpack cavity (Btu/lbm)

h_2 = Enthalpy of condensate leaving the overpack cavity (Btu/lbm)

m_c = Average rate of condensate flow measured during thermal equilibrium conditions (lbm/hr)

Based on the HI-STAR 100 overpack thermal model, a design basis minimum heat rejection capacity (Q_{hd}) shall be computed at the measured test conditions (i.e., steam temperature in the overpack cavity and ambient air temperature). The thermal test shall be considered acceptable if the measured heat rejection capability is greater than the design basis minimum heat rejection capacity ($Q_{hm} > Q_{hd}$).

The summary of reference ambient inputs that define the thermal test environment are provided in Table 9.1.4. In Figure 9.1.3, a steady-state temperature contour plot of a steam heated overpack is provided based on the thermal analysis methodology described in SAR Chapter 3. Transient heating of the overpack is also determined to establish the time required to approach (within 2° F (1° C)) the equilibrium temperatures. The surface temperature plot shown in Figure 9.1.4 demonstrates that a 12-hour steam heating time is adequate to closely approach the equilibrium condition.

If the acceptance criteria above are not met, then the HI-STAR 100 Package shall not be accepted until the root cause is determined, appropriate corrective actions are completed, and the overpack is re-tested with acceptable results.

Test results shall be documented and shall become part of the quality record documentation package.

9.1.7 Cask Identification

Each HI-STAR 100 System shall be *marked with a model number, identification number (to provide traceability back to documentation), and empty weight of the item in accordance with the marking requirements specified in* 10CFR72.236(k).

FIGURE 9.1.1

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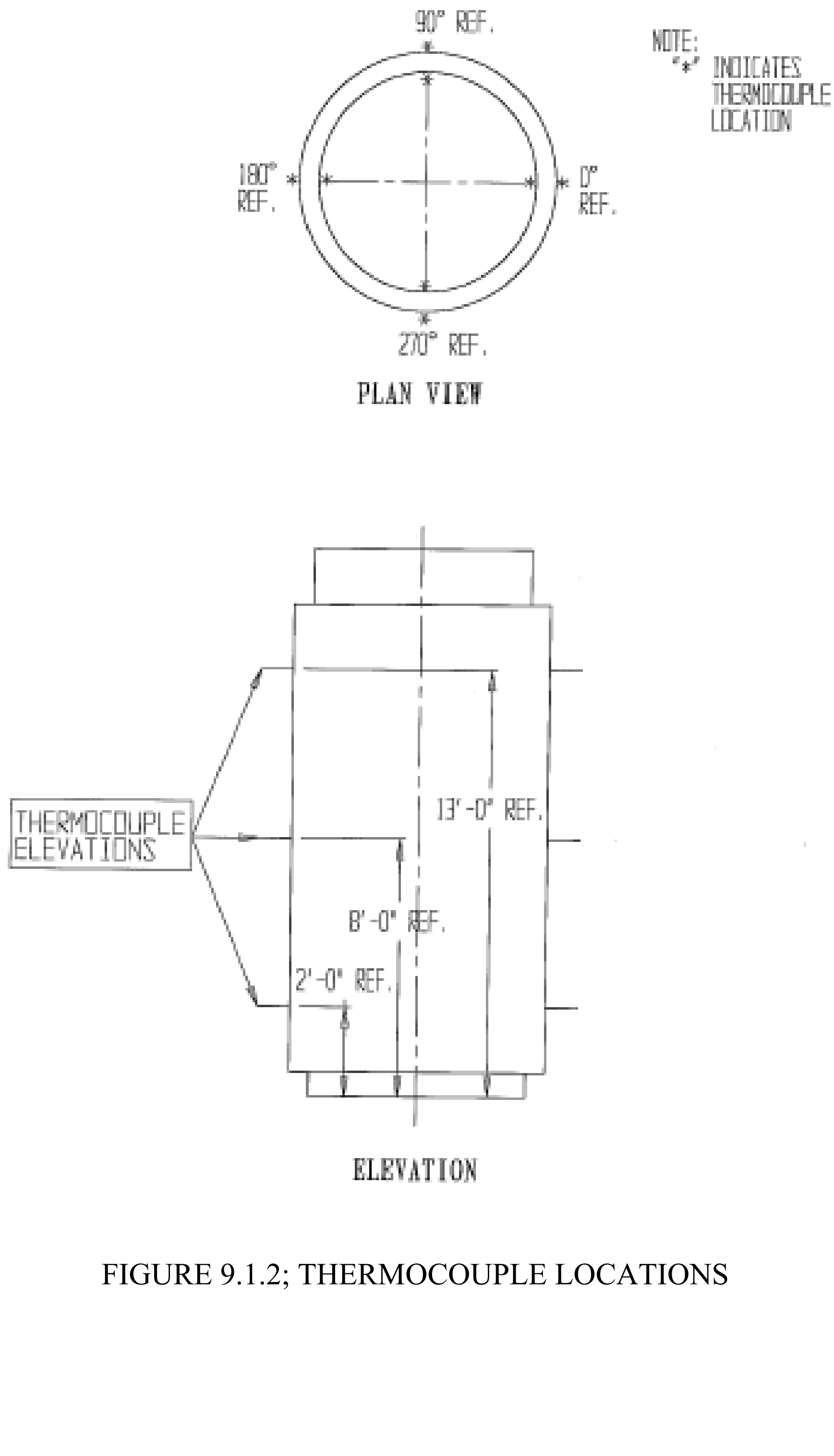

FIGURE 9.1.2; THERMOCOUPLE LOCATIONS

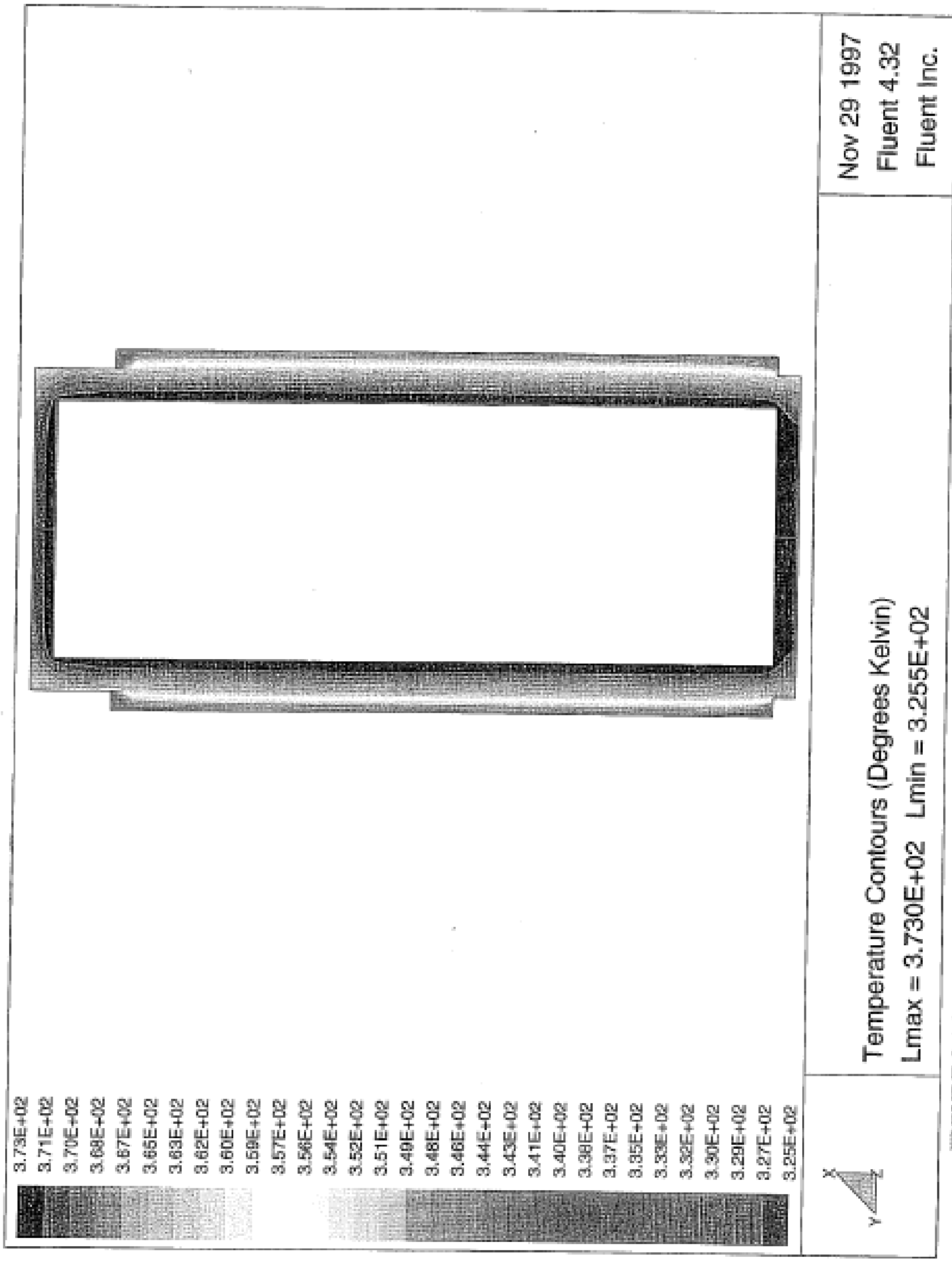

FIGURE 9.1.3: STEAM HEATED OVERPACK TEST CONDITION TEMPERATURE CONTOURS PLOT

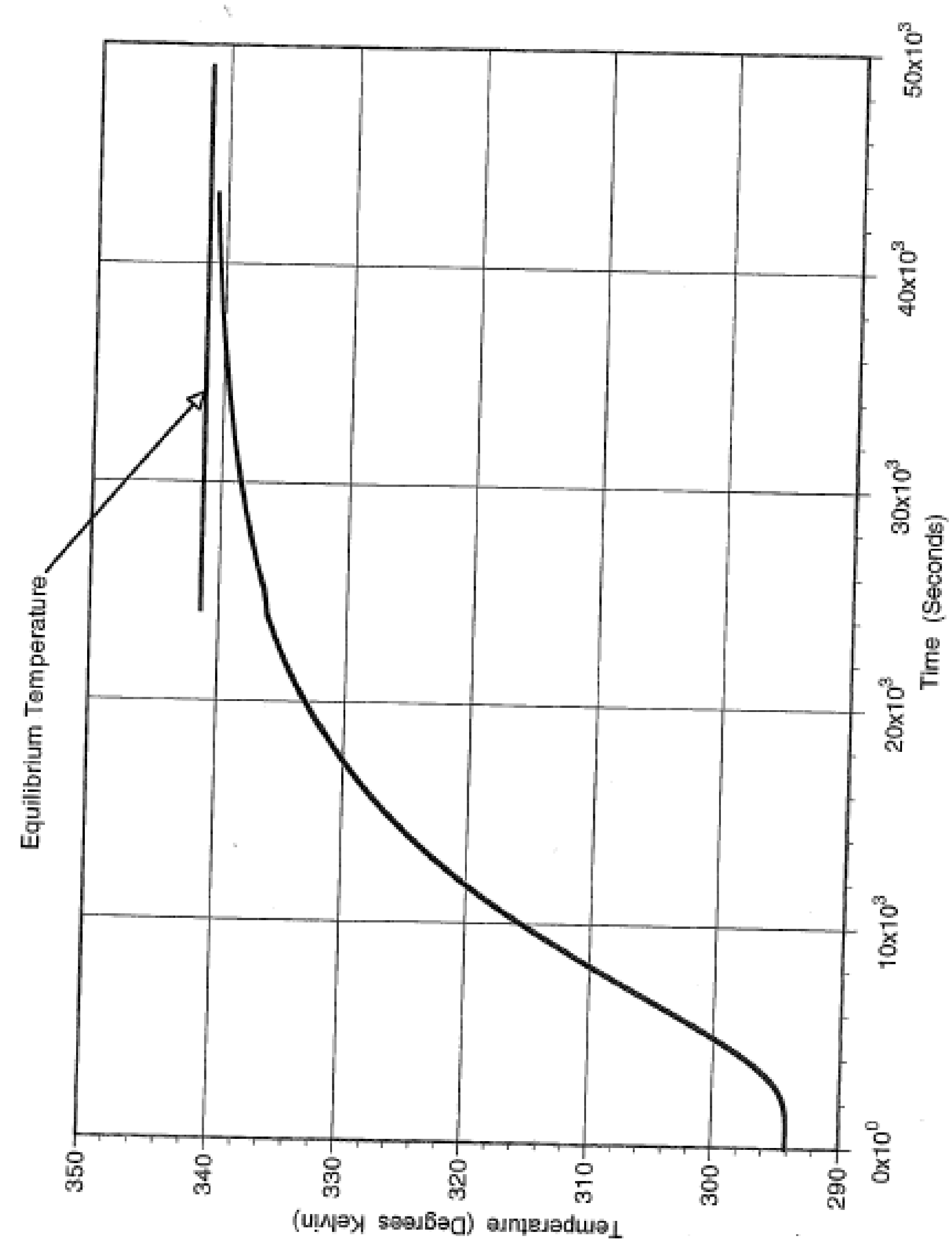

FIGURE 9.1.4: OVERPACK SURFACE TEMPERATURE HISTORY DURING A STEAM HEATED TEST

Table 9.1.1 *MPC INSPECTION AND TEST ACCEPTANCE CRITERIA*			
Function	***Fabrication***	***Pre-operation***	***Maintenance and Operations***
Visual Inspection and Nondestructive Examination (NDE)	*a) Assembly and examination of MPC components per ASME Code Section III, Subsections NB, NF, and NG, as defined on design drawings, per NB-5300, NF-5300, and NG-5300, as applicable.* *b) A dimensional inspection of the internal basket assembly and canister will be performed to verify compliance with design requirements.* *c) A dimensional inspection of the MPC lid and MPC closure ring will be performed prior to inserting into the canister shell to verify compliance with design requirements.* *d) NDE of weldments are defined on the design drawings using standard American Welding Society NDE symbols and/or notations.* *e) Cleanliness of the MPC shall be verified upon completion of fabrication.* *f) The packaging of the MPC at the completion of fabrication shall be verified prior to shipment.*	*a) The MPC shall be visually inspected prior to placement in service at the licensee's facility.* *b) MPC protection at the licensee's facility shall be verified.* *c) MPC cleanliness and exclusion of foreign material shall be verified prior to placing in the spent fuel pool.*	*a) None.*

Table 9.1.1 (continued) MPC INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Structural	a) Assembly and welding of MPC components *shall* be performed per ASME Code, Subsections NB, NF, and NG, as applicable. b) Materials analysis (steel, *neutron absorber, etc.), shall be performed and records shall* be kept in a manner commensurate with "important to safety" classifications.	a) None.	a) Multi-layer liquid penetrant (PT) examination of the MPC lid-to-shell weld shall be performed per ASME Section V, Article 5 (or ASME Section V, Article 2). Acceptance criteria for the examination are defined Table 9.1.3 and in the Design Drawings. b) ASME Code NB-6000 *pressure* test shall be performed after MPC closure welding. Acceptance criteria are defined in Section 9.1.2.2.2.
Leak Tests	*a) Helium leakage testing of the MPC shell and MPC shell to baseplate welds are performed on the unloaded MPC.* b) *Helium leakage testing of the MPC base metals (shell, baseplate, lid) is performed.*	a) None.	a) *Helium leak rate testing shall be performed on the vent and drain port cover plate to MPC lid field welds and the cover plate base metals.* Acceptance criteria are defined in the Technical Specifications.

Table 9.1.1 (continued) MPC INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Criticality Safety	a) The boron content *shall* be verified at the time of neutron absorber material manufacture. b) The installation of *neutron absorber* panels into MPC basket plates *shall* be verified by inspection.	a) None.	a) None.
Shielding Integrity	a) Material compliance shall be verified through CMTRs. b) Dimensional verification of MPC lid thickness shall be performed.	a) None.	a) None.
Thermal Acceptance	a) None.	a) None.	a) None.
Fit-up Tests	a) Fit-up of the following components is to be tested during fabrication. - MPC lid - vent/drain port cover plates - MPC closure ring b) A gauge test of all basket fuel compartments.	a) Fit-up of the following components *shall* be verified during pre-operation. - MPC lid - MPC closure ring - vent/drain cover plates	a) None.
Canister Identification Inspections	a) Verification of identification marking applied at completion of fabrication.	a) Identification marking *shall* be checked for legibility during pre-operation.	a) None.

Table 9.1.2
HI-STAR OVERPACK
INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	a) Assembly and examination *shall* be performed per ASME Code, Subsection NB, NB-5300 for helium retention boundary and Subsection NF, NF-5300 for non-helium retention boundary components. b) A dimensional inspection of the overpack internal cavity, external dimensions, and closure plate *shall* be performed to verify compliance with design requirements. c) NDE of weldments *shall* be defined on design drawings using standard American Welding Society NDE symbols and/or notations. d) Cleanliness of the HI-STAR overpack shall be verified upon completion of fabrication. e) Packaging of the HI-STAR overpack at the completion of fabrication *shall* be verified prior to shipment.	a) The HI-STAR overpack *shall* be visually inspected prior to placement in service at the licensee's facility. b) HI-STAR overpack protection at the licensee's facility *shall* be verified. c) HI-STAR overpack cleanliness and exclusion of foreign material shall be verified prior to use.	a) None.

Table 9.1.2 (continued)
HI-STAR OVERPACK
INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Structural	a) Assembly and welding of HI-STAR overpack components *shall* be performed per ASME Code, Subsection NB and NF, as applicable. b) Verification of structural materials *shall* be performed through receipt inspection and review of certified material test reports (CMTRs) obtained in accordance with the item's quality classification category. c) A load test of the lifting trunnions *shall* be performed during fabrication per ANSI N14.6. d) A pressure test of the helium retention boundary in accordance with ASME Code Section III, Subsection NB-6000 *shall* be performed. e) A pneumatic pressure test of the neutron shield enclosure *shall* be performed during fabrication.	a) None.	a) The rupture disks on *the neutron* shield vessel will be replaced every 5 years.

Table 9.1.2 (continued) HI-STAR OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Leak Tests	a) Helium leakage rate testing of the HI-STAR overpack helium retention boundary welds (e.g., containment boundary) *shall* be performed in accordance with ANSI N14.5. b) A fabrication verification helium leakage rate test shall be performed on all HI-STAR overpack mechanical seal boundaries in accordance with ANSI N14.5.	a) None.	a) Containment Fabrication Verification Leakage Tests of the HI-STAR 100 System shall be performed prior to commencement of transport operations.
Criticality Safety	a) None.	a) None.	a) None.
Shielding Integrity	a) Material verifications (Holtite-A, shell plates, etc.), *shall* be performed in accordance with the item's quality category. The required material certifications will be obtained. b) The placement of Holtite-A *shall* be controlled through written special process procedures.	a) None.	a) A shielding effectiveness test *shall* be performed after the first fuel loading and re-performed as specified in Table 9.2.1 while in service.

Table 9.1.2 (continued) HI-STAR OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operation**
Thermal Acceptance	a) A thermal acceptance test is performed on the first system, *at* completion of fabrication to confirm the heat transfer capabilities of the HI-STAR overpack.	a) None.	a) A thermal performance test of the HI-STAR 100 System shall be performed prior to commencement of transport operations.
Cask Identification Inspection	a) Identification plates *shall* be installed on the HI-STAR overpack at completion of the acceptance test program.	a) The identification plates will be checked prior to loading.	a) The identification plates *shall* be periodically inspected per licensee procedures and *shall* be repaired or replaced if damaged.
Functional Performance Tests	a) Fit-up tests of HI-STAR overpack components (closure plates, port plugs, *and cover* plates) *shall* be performed during fabrication.	a) Fit-up test of the *HI-STAR overpack* lifting trunnions with the lifting yoke *shall* be performed. b) Fit-up test of the HI-STAR overpack rotation trunnions with the horizontal transfer skid (if used) *shall* be performed. c) Fit-up test of the MPC into the HI-STAR overpack *shall* be performed prior to loading.	a) None.

Table 9.1.3
HI-STAR 100 NDE REQUIREMENTS

MPC			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
Shell longitudinal seam	RT PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 PT: ASME Section III, Subsection NB, Article NB-5350
Shell circumferential seam	RT PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 PT: ASME Section III, Subsection NB, Article NB-5350
Baseplate-to-shell	RT or UT PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 5 (UT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 UT: ASME Section III, Subsection NB, Article NB-5330 PT: ASME Section III, Subsection NB, Article NB-5350

Table 9.1.3 (continued) HI-STAR 100 NDE REQUIREMENTS			
MPC			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
Lid-to-shell	PT (root and final pass) PT (surface following hydrostatic test)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring-to-shell	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring-to-lid	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring radial welds	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Port cover plates-to-lid	PT (root and final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Lift lug and lift lug baseplate	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NG, Article NG-5350
Vent and drain port cover plate plug welds	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350

Table 9.1.3 (continued)
HI-STAR 100 NDE REQUIREMENTS

HI-STAR OVERPACK			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
Inner shell-to-top flange	RT MT or PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 7 (MT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 MT: ASME Section III, Subsection NB, Article NB-5340 PT: ASME Section III, Subsection NB, Article NB-5350
Inner shell-to-bottom plate	RT MT or PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 7 (MT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 MT: ASME Section III, Subsection NB, Article NB-5340 PT: ASME Section III, Subsection NB, Article NB-5350
Inner shell longitudinal seam	RT MT or PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 7 (MT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 MT: ASME Section III, Subsection NB, Article NB-5340 PT: ASME Section III, Subsection NB, Article NB-5350

Table 9.1.3 (continued) HI-STAR 100 NDE REQUIREMENTS			
HI-STAR OVERPACK			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
Inner shell circumferential seam	RT MT or PT (surface)	ASME Section V, Article 2 (RT) ASME Section V, Article 7 (MT) ASME Section V, Article 6 (PT)	RT: ASME Section III, Subsection NB, Article NB-5320 MT: ASME Section III, Subsection NB, Article NB-5340 PT: ASME Section III, Subsection NB, Article NB-5350
Intermediate shell welds (as noted on Design Drawings)	MT or PT (surface)	ASME Section V, Article 7 (MT) ASME Section V, Article 6 (PT)	MT: ASME Section III, Subsection NF, Article NF-5340 PT: ASME Section III, Subsection NF, Article NF-5350

Table 9.1.4

SUMMARY OF OVERPACK THERMAL ANALYSIS
AMBIENT INPUTS FOR STEAM HEATED TEST CONDITIONS

PARAMETER	VALUE
Steam Temperature	212ºF (100 ºC)
Ambient Temperature	70ºF(21.1 ºC)
Radiative Blocking	None
Exposed Surfaces Insolation	None

Table 9.1.5

RUPTURE DISK DATA

PARAMETER	VALUE
Relief Pressure	30 +/- 5 psig (207 +/- 35 kPA)
Pneumatic Pressure Test	37.5 +2.5 -0 psig (259 +17 -0 kPA) (125% of rupture disk relief set pressure)

9.2 MAINTENANCE PROGRAM

An ongoing maintenance program *shall* be defined and incorporated into the HI-STAR 100 System Operations Manual which *shall* be prepared and issued prior to the delivery and first use of the system *to each user*. This document *shall* delineate the detailed inspections, testing, and parts replacement necessary to ensure continued *structural, thermal and confinement performance;* radiological safety, *and* proper handling of the system in accordance with 10CFR72 [9.0.1] regulations, the conditions specified in the Certificate of Compliance, and the design requirements and criteria contained in this FSAR.

The HI-STAR 100 System is totally passive by design. There are no active components or monitoring systems required to assure the continued performance of its safety functions. As a result, only minimal maintenance will be required over the HI-STAR 100 System's lifetime, and this maintenance would primarily result from the weathering effects on the exterior coating system while in storage. Typical of such maintenance would be the reapplication of corrosion inhibiting materials on accessible external surfaces. Such maintenance requires methods and procedures no more demanding than those currently in use at licensed facilities.

Maintenance activities shall be performed under the licensee's NRC-approved quality assurance program. Maintenance activities shall be administratively controlled and the results documented. The maintenance program schedule for the HI-STAR 100 System is provided in Table 9.2.1.

9.2.1 Structural and Pressure Parts

Prior to each fuel loading, a visual examination in accordance with written and approved procedures *shall* be performed on the lifting trunnions (area outside of the overpack) and pocket trunnion recesses*, if used*. The examination *shall* inspect for indications of overstress such as cracking, deformation, or wear marks. Repairs or replacement in accordance with written and approved procedures *shall* be required if unacceptable conditions are identified.

As described in *FSAR* Chapters 7 and 11, there are no credible normal, off-normal, or accident events which can cause the structural failure of the MPC or HI-STAR overpack. Therefore, periodic structural or pressure tests on the MPCs or HI-STAR overpack following the initial acceptance tests are not required as part of the storage maintenance program.

9.2.2 Leakage Tests

There are no seals or gaskets that comprise the MPC confinement boundary since the MPC lid, port cover plates, and closure ring are welded closures. Metallic seals are used on the overpack helium retention boundary to ensure the retention of the helium in the overpack. These seals are not temperature sensitive within the design temperature range, are resistant to corrosion and radiation environments, and are helium leak tested after fuel loading. There are no credible normal, off-normal, or accident events which can cause the failure of the MPC confinement boundary or overpack helium retention boundary seals or welds. No leakage tests are required as part of the storage maintenance program.

Prior to transport of the HI-STAR 100 System following completion of the storage period, a Containment Periodic Verification leakage test shall be performed in accordance with ANSI N14.5 [9.1.9] and the HI-STAR 100 Safety Analysis Report [9.1.4] to verify the continued integrity of the containment boundary metallic seals.

9.2.3 Subsystem Maintenance

The HI-STAR 100 System does not include any subsystems which provide auxiliary cooling. Normal maintenance and calibration testing will be required on the vacuum drying, helium backfill, and leakage testing systems. Rigging, remote welders, cranes, and lifting beams shall also be inspected *prior to each loading campaign* to ensure proper maintenance and continued performance is achieved. Auxiliary shielding provided during on-site transfer operations or installed with the HI-STAR 100 at the storage pad requires no maintenance.

9.2.4 Rupture disks

The rupture disks shall be replaced *as described in Table 9.2.1* with approved spares per written and approved procedures.

9.2.5 Shielding

The gamma and neutron shielding materials in the overpack and MPC degrade negligibly over time or as a result of usage. To ensure continuing compliance of the HI-STAR 100 System to the design basis dose rate values, the Shielding Effectiveness Test shall be reperformed *as described in Table 9.2.1* after placement into service.

Radiation monitoring of the ISFSI by the licensee *in accordance with 10CFR72.104(c)* provides ongoing evidence and confirmation of the shielding integrity and performance. If increased radiation doses are indicated by the facility monitoring program, additional surveys of overpacks *shall* be performed to determine the cause of the increased dose rates.

The *neutron absorber* panels installed in the MPC baskets are not expected to degrade under normal long-term dry storage conditions. The use of Boral in similar nuclear applications is discussed in Chapter 1, and the long-term performance in a dry, inert gas atmosphere is evaluated in Chapter 3. *A similar discussion is provided for METAMIC ® neutron absorber material.* Therefore, no periodic verification testing of neutron poison material is required on the HI-STAR 100 System.

9.2.6 Thermal

There are no active cooling systems required for the long-term thermal performance of the HI-STAR 100 System. Therefore, no periodic thermal testing is required for the HI-STAR 100 System.

Table 9.2.1

HI-STAR 100 SYSTEM MAINTENANCE PROGRAM SCHEDULE

Task	Frequency
Overpack cavity visual inspection	Prior to fuel loading
Overpack bolt visual inspection	Prior to installation during each use
Overpack external surface (accessible) visual examination	Annually
HI-STAR 100 System Shield Effectiveness Test	After loading and every 20 years
Lifting trunnion and pocket trunnion recess visual inspection	Prior to next handling operation after loaded HI-STAR 100 System is placed on ISFSI pad.
Closure plate seal replacement	Following removal of closure plate bolting
Port seal replacement	Following opening of applicable port
Port cover plate seal replacement	Following removal of applicable cover plate
Replace neutron shield vessel rupture disks	Every 5 years

9.3 REGULATORY COMPLIANCE

Chapter 9 of this FSAR has been prepared to summarize the commitments of Holtec International to design, construct, and test the HI-STAR 100 System in accordance with the Codes and Standards identified in Chapter 2. Completion of the defined acceptance test program for each HI-STAR 100 System will provide assurance that the SSCs important to safety will perform their design function. The performance of the maintenance program by the licensee for each loaded HI-STAR 100 System will provide assurance for the continued safe long-term storage of the stored SNF.

The described acceptance criteria and maintenance programs can be summarized in the following evaluation statements:

1. Section 9.1 of this FSAR describes Holtec International's proposed program for preoperational testing and initial operations of the HI-STAR 100 System. Section 9.2 describes the proposed HI-STAR 100 maintenance program.

2. Structures, systems, and components (SSCs) of the HI-STAR 100 System designated as important to safety will be designed, fabricated, erected, assembled, inspected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. Tables 2.2.6 and 8.1.4 of this FSAR identify the safety importance and quality classifications of SSCs of the HI-STAR 100 System, and Tables 2.2.6 and 2.2.7 present the applicable standards for their design, fabrication, and inspection.

3. Holtec International will examine and test the HI-STAR 100 System to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness. Section 9.1 of this FSAR describes the MPC confinement boundary assembly, inspection, and testing.

4. Holtec International will mark the cask with a data plate indicating its model number, unique identification number, and empty weight.

5. It can be concluded that the acceptance tests and maintenance program for the HI-STAR 100 System are in compliance with 10CFR72 [9.0.1], and that the applicable acceptance criteria have been satisfied. The acceptance tests and maintenance program will provide reasonable assurance that the HI-STAR 100 System will allow safe storage of spent fuel throughout its certified term. This can be concluded based on a review that considers the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

9.4 REFERENCES

[9.0.1] U.S. Code of Federal Regulations, Title 10, "Energy", Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

[9.0.2] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", January 1997.

[9.1.1] American Society of Mechanical Engineers, " Boiler and Pressure Vessel Code," Sections II, III, V, IX, and XI, 1995, including Addenda through 1997.

[9.1.2] U.S. Code of Federal Regulations, Title 10, "Energy", Part 71, "Packaging and Transportation of Radioactive Material."

[9.1.3] American Society for Nondestructive Testing, "Personnel Qualification and Certification in Nondestructive Testing," Recommended Practice No. SNT-TC-1A, December 1992.

[9.1.4] HI-STAR 100 Safety Analysis Report, Holtec Report No. HI-951251, current revision.

[9.1.5] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kilograms) or More", ANSI N14.6, September 1993.

[9.1.6] NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", U.S. Nuclear Regulatory Commission, Washington, D.C., July 1980.

[9.1.7] U.S. Nuclear Regulatory Commission, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1m)," Regulatory Guide 7.11, June 1991.

[9.1.8] U.S. Nuclear Regulatory Commission, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 Inches (0.1m) But Not Exceeding 12 Inches (0.3m)," Regulatory Guide 7.12, June 1991.

[9.1.9] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials Leakage Tests on Packages for Shipment", ANSI N14.5-1997.

[9.1.10] Holtec International Position Paper DS-213, “Acceptable Flaw Size in MPC Lid-to-Shell Welds”, Revision 2.

CHAPTER 10: RADIATION PROTECTION

This chapter discusses the design considerations and operational features that are incorporated in the HI-STAR 100 System design to protect plant personnel and the public from exposure to radioactive contamination and ionizing radiation during canister loading, closure, on-site movement, and on-site dry storage. Occupational exposure estimates for typical MPC loading, closure, on-site movement operations, and ISFSI inspections are provided. An off-site dose assessment for a typical ISFSI is also discussed. Since the determination of off-site doses is necessarily site-specific, similar dose assessments are to be prepared by the licensee, as part of implementing the HI-STAR 100 System in accordance with 10CFR72.212 [10.0.1]. The information provided in this chapter is in full compliance with the requirements of NUREG-1536 [10.0.2].

Much of the new information in this chapter is directly extracted from the HI-STORM 100 FSAR [10.0.3] and HI-STORM FW FSAR [10.0.4] this information is shown in *italics*. All changes in this revision are marked with revision bars[1].

10.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS-LOW-AS-REASONABLY-ACHIEVABLE (ALARA)

10.1.1 Policy Considerations

The HI-STAR 100 System has been designed in accordance with 10CFR72 [10.0.1] and maintains radiation exposures ALARA consistent with 10CFR20 [10.1.1] and the guidance provided in Regulatory Guides 8.8 [10.1.2] and 8.10 [10.1.3]. Licensees using the HI-STAR 100 System will utilize and apply their existing site ALARA policies, procedures and practices for ISFSI activities to ensure that personnel exposure requirements of 10CFR20 [10.1.1] are met. Personnel performing ISFSI operations shall be trained on the operation of the HI-STAR 100 System, and be familiarized with the expected dose rates around the MPC and overpack during all phases of loading, storage, and unloading operations. Chapter 12 provides dose rate limits for the MPC lid and the overpack surfaces to ensure that the HI-STAR 100 System is operated within design basis conditions and that ALARA goals will be met. Pre-job ALARA briefings should be held with workers and radiological protection personnel prior to work on or around the system. Worker dose rate monitoring, in conjunction with trained personnel and well-planned activities, will significantly reduce the overall dose received by the workers. When preparing or making changes to site-specific procedures for ISFSI activities, users shall ensure that ALARA practices are implemented and the 10CFR20 [10.1.1] standards for radiation protection are met in accordance with the site's written commitments. Users *can* further reduce dose rates around the HI-STAR 100 System by preferentially loading longer-cooled and lower-burnup spent fuel assemblies in the periphery fuel storage cells of the MPC, and loading assemblies with shorter cooling times and higher burnups in the inner MPC fuel storage cell locations as specified in the

[1] Wherever multiple units are shown, the US units are the governing value, and the SI units are shown for information only.

Technical Specifications. Users can also further reduce the dose rates around the HI-STAR 100 System by the use of temporary shielding. Temporary shielding is discussed in Section 10.1.4.

10.1.2 Radiation Exposure Criteria

The radiological protection criteria that limit exposure to radioactive effluents and direct radiation from an ISFSI using the HI-STAR 100 System are as follows:

1. 10CFR72.104 [10.0.1] requires that for normal operation and anticipated occurrences, the annual dose equivalent to any real individual located beyond the owner-controlled area boundary must not exceed 25 mrem (0.25 mSv) to the whole body, 75 mrem (0.75 mSv) to the thyroid, and 25 mrem (0.25 mSv) to any other *critical* organ. This dose would be a result of planned discharges, direct radiation from the ISFSI, and any other radiation from uranium fuel cycle operations in the area. The licensee is responsible for demonstrating site-specific compliance with these requirements.

2. 10CFR72.106 [10.0.1] requires that any individual located on or beyond the nearest owner-controlled area boundary *may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 rem (50 mSv), or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 rem (500 mSv). The lens dose equivalent may not exceed 15 rem (150 mSv) and the shallow dose equivalent to skin or any extremity may not exceed 50 rem (500 mSv).* The licensee is responsible for demonstrating site-specific compliance with this requirement.

3. 10CFR20 [10.1.1], Subparts C and D, limit occupational exposure and exposure to individual members of the public. The user is responsible for demonstrating site-specific compliance with this requirement.

4. Regulatory Position 2 of Regulatory Guide 8.8 [10.1.2] provides guidance regarding facility and equipment design features. This guidance has been followed in the design of the HI-STAR 100 System as described below:

    - Regulatory Position 2a, regarding access control, is met by locating the ISFSI in a Protected Area in accordance with 10CFR72.212(b)(5)(ii) [10.0.1]. *Depending on the site-specific design, other equivalent measures may be used.* Unauthorized access is prevented once a loaded HI-STAR 100 System is placed in an ISFSI. Due to the nature of the system, only limited monitoring is required, thus reducing occupational exposure and supporting ALARA considerations. The licensee is responsible for site-specific compliance with these criteria.

    - Regulatory Position 2b, regarding radiation shielding, is met by the overpack biological shielding that minimizes personnel exposure as described in Chapter 8. Fundamental design considerations that most directly influence occupational exposures with dry storage systems in general and which have been incorporated into the HI-STAR 100 System design include:

- system designs that reduce or minimize the number of handling and transfer operations for each spent fuel assembly;
- system designs that reduce or minimize the number of handling and transfer operations for each MPC loading;
- system designs that maximize fuel capacity, thereby taking advantage of the self-shielding characteristics of the fuel and the reduction in the number of MPCs that must be loaded and handled;
- *system designs that minimize planned maintenance requirements*;
- system designs that minimize decontamination requirements at ISFSI decommissioning;
- system designs that optimize the placement of shielding with respect to anticipated worker locations and fuel placement;
- thick-walled overpacks that provide gamma and neutron shielding;
- thick MPC lid which provides effective shielding for operators during MPC loading and unloading operations;
- multiple welded barriers to confine radionuclides;
- smooth surfaces to reduce decontamination time;
- minimization of potential crud traps on the handling equipment to reduce decontamination requirements;
- capability of maintaining water in the MPC and annulus during MPC welding to reduce dose rates;
- capability of maintaining water in the annulus space to reduce dose rates during closure operations;
- MPC penetrations located and configured to reduce streaming paths;
- overpack penetrations located and oriented to reduce streaming paths;
- MPC vent and drain ports with re-sealable caps to prevent the release of radionuclides during loading and unloading operations and facilitate draining, drying, and backfill operations;
- use of an annulus seal and annulus overpressure system to prevent contamination of the MPC shell outer surfaces during in-pool activities;
- available temporary and auxiliary shielding to reduce dose rates around the overpack; and
- low-maintenance design to reduce doses during storage operation.

- Regulatory Position 2c, regarding process instrumentation and controls, is met since there are no radiation instrumentation and controls needed at the ISFSI.

- Regulatory Position 2d, regarding control of airborne contaminants, is met since the HI-STAR 100 System is designed to withstand all design basis conditions without loss of confinement function, as described in Chapter 7 of this FSAR, and no gaseous releases are anticipated. No significant surface contamination is expected since the exterior of the MPC is kept clean by using clean water in the overpack-MPC annulus and by using an inflatable annulus seal and optional annulus overpressure system.

- Regulatory Position 2e, regarding crud control, is not applicable to a HI-STAR 100 System ISFSI since there are no radioactive systems at an ISFSI that could transport crud.

- Regulatory Position 2f, regarding decontamination, is met since the exterior of the loaded overpack is decontaminated prior to being removed from the plant's fuel building. The exterior surface of the overpack is designed for ease of decontamination. In addition, an inflatable annulus seal and optional annulus overpressure system is used to prevent fuel pool water from contacting and contaminating the exterior surface of the MPC.

- Regulatory Position 2g, regarding monitoring *of airborne radioactivity*, is met since the *MPC provides confinement for all design basis conditions. There is no need for monitoring since no airborne radioactivity is anticipated to be released from the casks at an ISFSI*;

- Regulatory Position 2h, regarding resin treatment systems, is not applicable to an ISFSI since there are no treatment systems containing radioactive resins.

- Regulatory Position 2i, regarding other miscellaneous ALARA items, is met since stainless steel is used in the MPC shell, the primary confinement boundary. This material is resistant to the damaging effects of radiation and is well proven in cask use. Use of this material quantitatively reduces or eliminates the need to perform maintenance (or replacement) on the primary confinement system.

10.1.3 Operational Considerations

Operational considerations that most directly influence occupational exposures with dry storage systems in general and that have been incorporated into the design of the HI-STAR 100 System include:

- totally-passive design requiring minimal maintenance and monitoring (other than security monitoring) during storage;

- remotely operated welding system, lift yoke, weld removal system and *moisture removal systems* to reduce time operators spend in the vicinity of the loaded MPC;

- maintaining water in the MPC and the annulus region during MPC closure activities to reduce dose rates;

- descriptive operating procedures that provide guidance to reduce equipment contamination, obtain survey information, minimize dose and alert workers to possible changing radiological conditions;

- preparation and inspection of the overpack in low-dose areas;

- MPC lid fit tests and inspections prior to actual loading to ensure smooth operation during loading;

- gas sampling of the MPC and HI-STAR 100 System annulus (receiving from transport) to assess the condition of the cladding and MPC confinement boundary prior to opening;

- fuel cool-down operations developed for fuel unloading operations which minimize thermal shock to the fuel and therefore reduce the potential for fuel cladding rupture;

- wetting of component surfaces prior to placement in the spent fuel pool to reduce the need for decontamination;

- decontamination practices which consider the effects of weeping during overpack heat up and surveying of the overpack prior to removal from the fuel handling building;

- incorporation of ALARA principles in operation, surveillance, and maintenance procedures;

- a sequence of operations based on ALARA considerations; and

- use of mock-ups to prepare personnel for actual work situations.

10.1.4 Auxiliary/Temporary Shielding

To minimize occupational and site boundary doses, the HI-STAR 100 System has optional auxiliary shielding available for use during loading, storage and unloading operations. The HI-STAR 100 System auxiliary shielding consists of the Automated Welding System Baseplate, the overpack temporary shield ring, the annulus shield, the overpack bottom cover, the pocket trunnion neutron shield plugs, and the overpack bottom ring shield. Each auxiliary shield is described in Table 10.1.1, and the procedures for utilization are provided in Chapter 8. Users shall evaluate the need for auxiliary and temporary shielding based on an ALARA review of each loading operation. For fuel assemblies with lower burnups and longer cooling times, the need for auxiliary and temporary shielding is reduced.

Table 10.1.1
HI-STAR 100 System AUXILIARY AND TEMPORARY SHIELDS

Temporary Shield	Description	Utilization
Automated Welding System Baseplate	Thick gamma and neutron shield circular plate that sits on the MPC lid. Plate is set directly on the MPC lid. Threaded lift holes are provided to assist in rigging.	Used during MPC closure and unloading operations in the cask preparation area to reduce the dose rates around the MPC lid. The design of the closure ring allows the baseplate shield to remain in place during the entire closure operation.
Overpack Temporary Shield Ring	A shield that fits on the cask neutron shield around the upper forging and provides supplemental shielding to personnel performing cask loading and closure operations.	Used during MPC and overpack closure operations to reduce dose rates to the operators around the top flange of the overpack.
Annulus Shield	A shield that is seated between the MPC shell and the overpack.	Used during MPC closure operations to reduce streaming from the annulus.
Overpack Bottom Cover	A cup-shaped gamma and neutron shield cover that is attached to the overpack bottom and secured using the impact limiter bolt holes.	Used during on-site horizontal transfer of the loaded overpack to reduce dose rates from the bottom of the overpack.
Overpack Bottom Ring	A series of segmented, concrete rings that are placed under the neutron shield around the base of the overpack. The ring segments when positioned, form a complete ring around the overpack base. The rings are placed in position on the ISFSI pad and are not secured.	Used during storage of the overpacks on the ISFSI pad to reduce the dose rates around the base of the overpack.
Pocket Trunnion Neutron Shield Plugs	A custom-fit stainless steel clad neutron shielding material that is inserted and bolted into the pocket trunnions (if used).	Used during storage of the overpack on the ISFSI pad. Reduces the neutron dose rate around the optional pocket trunnions.

10.2 RADIATION PROTECTION DESIGN FEATURES IN THE SYSTEM DESIGN

The development of the HI-STAR 100 System has focused on design provisions to address the considerations summarized in Sections 10.1.2 and 10.1.3. The following specific design features ensure a high degree of confinement integrity and radiation protection:

- HI-STAR 100 System has been designed to meet storage condition dose rates required by 10CFR72 [10.0.1] containing spent fuel assemblies cooled at least 5 years;

- HI-STAR 100 System has been designed to accommodate a maximum number of PWR or BWR fuel assemblies to minimize the number of cask systems that must be handled and stored at the storage facility and later transported off-site;

- HI-STAR 100 System is low maintenance because of the outer metal shell. The metal shell and its protective coating are extremely resistant to degradation;

- HI-STAR 100 System has been designed for redundant, multi-pass welded closures on the MPC; consequently, no monitoring of the confinement boundary is necessary and no gaseous or particulate releases occur for normal, off-normal or postulated accident conditions; and

- HI-STAR 100 System has auxiliary shielding devices which eliminate streaming paths and simplify operations.

- *HI-STAR 100 has been designed to allow close positioning (pitch) on the ISFSI storage pad, thereby increasing the ISFSI self-shielding by decreasing the view factors and reducing exposures to on-site and off-site personnel*. For horizontal storage, the lateral distance between cask centerlines shall not be less than 12 feet (3.66 m) to provide practical access for handling equipment. As described in Chapter 4, a site-specific spacing evaluation is required for horizontal casks.

10.3 ESTIMATED ON-SITE COLLECTIVE DOSE ASSESSMENT

This section provides the estimates of the cumulative exposure to personnel performing loading and unloading operations using the HI-STAR 100 System. This section uses the shielding analysis provided in Chapter 5 and the operations procedures provided in Chapter 8 to develop a dose rate assessment for loading and unloading operations. The dose rate assessments are provided in Table 10.3.1 and Table 10.3.2 for loading and unloading operations, respectively.

The dose rates on and around the HI-STAR 100 System overpack and MPC lid are estimated using an 18-inch (35.7 cm), on-contact and 1-meter dose rates for the overpack during the loading and unloading operations. The dose rates around the overpack are based on 24 PWR fuel assemblies with a burnup of 40,000 MWD/MTU and cooling of 5 years. The selection of this fuel assembly type bounds all possible loading scenarios for the HI-STAR 100 System with MPC-24 from a dose-rate perspective. Similar results are expected for the HI-STAR 100 System with MPC-32 and MPC-68. No assessment is made with respect to radiation levels around the cask during operations where no fuel is in the MPC since radiation levels vary significantly by site and locations within. In addition, exposures are based on work being performed without the temporary shielding described in Section 10.1.4.

The dose rate location points around the overpack were selected to model actual worker locations. Cask operators typically work at an arms-reach distance from the cask. To account for this, either an 18-inch (35.7 cm) distance or a rough average of on-contact and 1-meter dose rates were used to roughly estimate the dose rate for the worker. This assessment takes credit for the actual number of workers directly working around the cask and the actual time spent in the vicinity of the cask. The duration times and number of workers are based on historical accounts of spent fuel canister loading operations at nuclear utilities, taking into account the proximity of controls and remote control features of the HI-STAR 100 ancillary equipment. For example, the Vacuum Drying System and Automated Welding System are remotely operated to minimize the amount of time the operators need to spend in direct contact with the cask. Typically, once the cask is configured for a specific task, the operators are free to exit the work area and continue operations from an ALARA low-dose area.

Table 10.3.1 provides a summary of the dose assessment for a HI-STAR 100 System loading operation. Table 10.3.2 provides a summary of the dose assessment for a HI-STAR 100 System unloading operation. Because of the various operational requirements for the different sites, a conservative approach on operations was used to assess the personnel exposures. The personnel requirements and anticipated duration of activities are based on previous utility canister loading experience and published data.

10.3.1 Estimated Exposures for Loading and Unloading Operations

The assumptions discussed above are conservative by design. Historically, actual occupational doses to load and place canister-based systems in storage are significantly lower than the projected values for those systems. The main factors attributed to the lower-than-projected personnel exposures are the age of the spent fuel, conservative assumptions in the dose estimates, and good ALARA practices. These same considerations are expected to factor into the actual operation of the HI-STAR 100 System. To estimate the dose received by a single worker, it should be understood that a canister-based system requires a diverse range of disciplines to perform all the necessary functions. Technical Specifications with time limits and control of utility restart conditions have prompted utilities to load canister systems in a round-the-clock mode. This results in the exposure being spread out over a team of operators and technicians with no single discipline receiving a majority of the exposure.

The dose rates provided in Tables 10.3.1 and 10.3.2 are conservatively based on fuel assemblies with 40,000 MWD/MTU and 5-year cooling which bounds the allowable burnup and cooling time combinations for the HI-STAR 100 System with MPC-24. Similar results are expected for the HI-STAR 100 System with MPC-32 and MPC-68. The total person-rem exposure from operation of the HI-STAR 100 System is proportional to the number of systems loaded. A typical utility will load approximately four MPCs per reactor cycle to maintain the current available spent fuel pool capacity. Utilities requiring dry storage of spent fuel assemblies typically have a large inventory of spent fuel assemblies that date back to the reactor's first cycle. The older fuel assemblies will have a significantly lower dose rate than the design basis fuel assemblies. Users shall assess the cask loading for their particular fuel types (age, burnup, cooling time) to satisfy the requirements of 10CFR20 [10.1.1].

10.3.2 Estimated Exposures for Surveillance and Maintenance

Table 10.3.3 provides the maximum anticipated occupational exposure received from security surveillance and maintenance of an ISFSI. Although the HI-STAR 100 System requires minimal maintenance during storage, maintenance will be required around the ISFSI for items such as security equipment maintenance, grass cutting, snow removal, drainage system maintenance, and lighting, telephone, and intercom repair. Security surveillance time is based on a daily security patrol around the perimeter of the ISFSI security fence. Users may opt to utilize remote security viewing methods instead of performing direct visual observation of the ISFSI. Since security surveillances can be performed from outside the ISFSI, a dose rate of 4 mrem/hour (0.04 mSv/hr) is conservatively used. The estimated dose rates described below are based on a sample array of HI-STAR 100 Systems fully loaded with design basis fuel assemblies, placed at their minimum required pitch, in a 2 x 6 HI-STAR 100 System array. The maintenance worker is assumed to be at a distance of 5 meters from the center of the long edge of the array. For maintenance of the casks and the ISFSI, a dose rate of 50 mrem/hour (0.5 mSv/hr) is estimated.

Table 10.3.1
HI-STAR 100 SYSTEM LOADING OPERATIONS
ESTIMATED OPERATIONAL EXPOSURES (40,000MWD/MTU, 5-YEAR COOLED FUEL)

ACTIVITY	NUMBER OF WORKERS†	DURATION (HOURS)††	ESTIMATED DOSE RATE (MREM/HR) (mSv/h)	OCCUPATIONAL DOSE TO INDIVIDUAL (PERSON-MREM) (PERSON-mSv)	ESTIMATED TOTAL DOSE FOR TASK (PERSON-MREM) (PERSON-mSv)
REMOVE HI-STAR CLOSURE PLATE	2	1	0 (0)	0 (0)	0 (0)
INSTALL EMPTY MPC	3	2	0 (0)	0 (0)	0 (0)
INSTALL UPPER FUEL SPACERS	3	4	0 (0)	0 (0)	0 (0)
INSTALL LOWER FUEL SPACERS	3	4	0 (0)	0 (0)	0 (0)
FILL MPC AND ANNULUS	2	4	0 (0)	0 (0)	0 (0)
INSTALL ANNULUS SEAL	1	0.3	0 (0)	0 (0)	0 (0)
PLACE HI-STAR IN SPENT FUEL POOL	3	1.2	5 (0.05)	6 (0.06)	18 (0.018)
LOAD FUEL ASSEMBLIES INTO MPC	3	11.3	5 (0.05)	56.5 (0.565)	170 (1.70)
PERFORM ASSEMBLY IDENTIFICATION VERIFICATION	3	1.5	5 (0.05)	7.5 (0.075)	22.5 (0.225)
INSTALL DRAIN LINE TO MPC LID	3	0.8	5 (0.05)	4 (0.04)	12 (0.12)
ALIGN MPC LID AND LIFT YOKE TO DRAIN LINE	2	0.2	5 (0.05)	1 (0.01)	2 (0.02)
INSTALL MPC LID	2	0.4	5 (0.05)	2 (0.02)	4 (0.04)
REMOVE HI-STAR FROM SPENT FUEL POOL	2	0.4	18.5 (0.185)	7.4 (0.074)	14.8 (0.148)
DECONTAMINATE HI-STAR BOTTOM	2	0.2	44 (0.44)	8.8 (0.088)	17.6 (0.176)
SET HI-STAR IN CASK PREPARATION AREA	2	0.5	20 (0.20)	10 (0.10)	20 (0.20)
MEASURE DOSE RATES AT MPC LID	1	0.2	18.5 (0.185)	3.7 (0.037)	3.7 (0.037)
DECONTAMINATE HI-STAR AND LIFT YOKE	3	0.7	20 (0.20)	14 (0.14)	42 (0.42)
INSTALL TEMPORARY SHIELD RING	2	0.3	22 (0.22)	6.6 (0.066)	13.2 (0.132)
REMOVE INFLATABLE ANNULUS SEAL	1	0.1	18.5 (0.185)	1.85 (0.0185)	1.85 (0.0185)

† Indicates number of workers in direct or close contact with HI-STAR 100.
†† Indicates actual duration of work in direct or close contact with HI-STAR 100.

Table 10.3.1 (Continued)
HI-STAR 100 SYSTEM LOADING OPERATIONS
ESTIMATED OPERATIONAL EXPOSURES (40,000MWD/MTU, 5-YEAR COOLED FUEL)

ACTIVITY	NUMBER OF WORKERS†	DURATION (HOURS)††	ESTIMATED DOSE RATE (MREM/HR) (mSv/h)	OCCUPATIONAL DOSE TO INDIVIDUAL (PERSON-MREM) (PERSON-mSv)	ESTIMATED TOTAL DOSE FOR TASK (PERSON-MREM) (PERSON-mSv)
LOWER ANNULUS WATER LEVEL SLIGHTLY	1	0.2	18.5 (0.185)	3.7 (0.037)	3.7 (0.037)
SMEAR MPC LID TOP SURFACES	1	0.2	18.5 (0.185)	3.7 (0.037)	3.7 (0.037)
INSTALL ANNULUS SHIELD	1	0.1	18.5 (0.185)	1.85 (0.0185)	1.85 (0.0185)
LOWER MPC WATER LEVEL	2	0.5	18.5 (0.185)	9.25 (0.0925)	18.5 (0.185)
WELD MPC LID & Perform NDE	2	1.2	18.5 (0.185)	22.2 (0.222)	44.4 (0.444)
PERFORM VOL EXAM OF MPC WELD	2	0.3	18.5 (0.185)	5.55 (0.0555)	11.1 (0.111)
RAISE MPC WATER LEVEL	2	0.1	18.5 (0.185)	1.85 (0.0185)	3.7 (0.037)
PERFORM HYDRO TEST ON MPC	2	0.3	18.5 (0.185)	5.55 (0.0555)	11.1 (0.111)
PERFORM LEAKAGE TESTING	2	0.5	18.5 (0.185)	9.25 (0.0925)	18.5 (0.185)
DRAIN MPC	1	0.7	77 (0.77)	53.9 (0.539)	53.9 (0.539)
MEASURE VOLUME OF WATER DRAINED	1	0.1	77 (0.77)	7.7 (0.077)	7.7 (0.077)
VACUUM DRY MPC	1	0.3	77 (0.77)	23.1 (0.231)	23.1 (0.231)
PERFORM MPC DRYNESS VERIFICATION TEST	2	0.1	77 (0.77)	7.7 (0.077)	15.4 (0.154)
BACKFILL MPC	2	0.2	77 (0.77)	15.4 (0.154)	30.8 (0.308)
WELD VENT AND DRAIN PORT COVER PLATES	1	0.2	77 (0.77)	15.4 (0.154)	15.4 (0.154)
PERFORM A LIQUID PENETRANT EXAMINATION	2	0.3	77 (0.77)	23.1 (0.231)	46.2 (0.462)
PERFORM LEAKAGE TEST ON COVER PLATES	2	0.2	77 (0.77)	15.4 (0.154)	30.8 (0.308)

† Indicates number of workers in direct or close contact with HI-STAR 100.
†† Indicates actual duration of work in direct or close contact with HI-STAR 100.

Table 10.3.1 (Continued)
HI-STAR 100 SYSTEM LOADING OPERATIONS
ESTIMATED OPERATIONAL EXPOSURES (40,000MWD/MTU, 5-YEAR COOLED FUEL)

ACTIVITY	NUMBER OF WORKERS†	DURATION (HOURS)††	ESTIMATED DOSE RATE (MREM/HR) (mSv/h)	OCCUPATIONAL DOSE TO INDIVIDUAL (PERSON-MREM) (PERSON-mSv)	ESTIMATED TOTAL DOSE FOR TASK (PERSON-MREM) (PERSON-mSv)
WELD MPC CLOSURE RING	1	0.4	77 (0.77)	30.8 (0.308)	30.8 (0.308)
PERFORM NDE ON CLOSURE RING WELDS	2	0.3	77 (0.77)	23.1 (0.231)	46.2 (0.462)
DRAIN ANNULUS	1	0.2	185 (1.85)	37 (0.37)	37 (0.37)
PERFORM SURVEYS ON HI-STAR	1	0.2	85 (0.85)	17 (0.17)	17 (0.17)
REMOVE ANNULUS SHIELD	1	0.1	77 (0.77)	7.7 (0.077)	7.7 (0.077)
INSTALL HI-STAR CLOSURE PLATE	3	1.5	17.6 (0.176)	26.4 (0.264)	79.2 (0.0.792)
VACUUM DRY HI-STAR ANNULUS	1	0.2	17.6 (0.176)	3.52 (0.0352)	3.52 (0.0352)
BACKFILL HI-STAR ANNULUS	1	0.2	17.6 (0.176)	3.52 (0.0352)	3.52 (0.0352)
LEAKTEST HI-STAR ANNULUS	2	0.5	73.4 (0.734)	36.7 (0.367)	73.4 (0.734)
REMOVE TEMPORARY SHIELD RING	2	0.2	93 (0.93)	18.6 (0.186)	37.2 (0.372)
PERFORM FINAL SURVEYS ON HI-STAR	1	0.2	85 (0.85)	17 (0.17)	17 (0.17)
PLACE HI-STAR IN STORAGE	2	1.3	85 (0.85)	110.5 (1.105)	221 (2.21)
INSTALL HI-STAR POCKET TRUNNION PLUGS	1	0.2	185 (1.85)	37 (0.37)	37 (0.37)
INSTALL BOTTOM SHIELD RING	2	0.2	185 (1.85)	37 (0.37)	74 (0.74)
TOTAL					1365.9 (13.659)

† Indicates number of workers in direct or close contact with HI-STAR 100.
†† Indicates actual duration of work in direct or close contact with HI-STAR 100.

Table 10.3.2
HI-STAR 100 SYSTEM UNLOADING OPERATIONS
ESTIMATED OPERATIONAL EXPOSURES (40,000MWD/MTU, 5-YEAR COOLED FUEL)

ACTIVITY	NUMBER OF WORKERS†	DURATION (HOURS)††	ESTIMATED DOSE RATE (MREM/HR) (mSv/h)	OCCUPATIONAL DOSE TO INDIVIDUAL (PERSON-MREM) (PERSON-mSv)	ESTIMATED TOTAL DOSE FOR TASK (PERSON-MREM) (PERSON-mSv)
REMOVE BOTTOM SHIELD RING	2	0.2	185 (1.85)	37 (0.37)	74 (0.74)
REMOVE HI-STAR POCKET TRUNNION PLUGS	1	0.2	185 (1.85)	37 (0.37)	37 (0.37)
RECOVER HI-STAR FROM STORAGE	2	1.3	85 (0.85)	110.5 (1.105)	221 (2.21)
PLACE HI-STAR IN DESIGNATED PREPARATION AREA	2	0.6	85 (0.85)	51 (0.51)	102 (1.02)
SAMPLE ANNULUS GAS	2	0.3	18 (0.18)	5.4 (0.054)	10.8 (0.108)
REMOVE HI-STAR CLOSURE PLATE	2	1	77 (0.77)	77 (0.77)	154 (1.54)
FILL ANNULUS	1	0.2	77 (0.77)	15.4 (0.154)	15.4 (0.154)
INSTALL ANNULUS SHIELD	1	0.1	77 (0.77)	7.7 (0.077)	7.7 (0.077)
REMOVE MPC CLOSURE RING	1	0.4	77 (0.77)	30.8 (0.308)	30.8 (0.308)
REMOVE VENT PORT COVERPLATE WELD AND SAMPLE MPC GAS	1	0.4	77 (0.77)	30.8 (0.308)	30.8 (0.308)
PERFORM MPC COOL-DOWN	1	0.2	77 (0.77)	15.4 (0.154)	15.4 (0.154)
FILL MPC CAVITY WITH WATER	1	0.7	77 (0.77)	53.9 (0.539)	53.9 (0.539)
REMOVE MPC LID TO SHELL WELD	1	0.7	18 (0.18)	12.6 (0.126)	12.6 (0.126)
INSTALL INFLATABLE SEAL	1	0.1	18 (0.18)	1.8 (0.018)	1.8 (0.018)
PLACE HI-STAR IN SPENT FUEL POOL	2	0.4	20 (0.20)	8 (0.08)	16 (0.16)
REMOVE MPC LID	2	0.4	5 (0.05)	2 (0.02)	4 (0.04)
REMOVE SPENT FUEL ASSEMBLIES FROM MPC	3	11.3	5 (0.05)	56.5 (0.565)	113 (1.13)

† Indicates number of workers in direct or close contact with HI-STAR 100.
†† Indicates actual duration of work in direct or close contact with HI-STAR 100.

Table 10.3.2 (Continued)
HI-STAR 100 SYSTEM UNLOADING OPERATIONS
ESTIMATED OPERATIONAL EXPOSURES (40,000MWD/MTU, 5-YEAR COOLED FUEL)

ACTIVITY	NUMBER OF WORKERS†	DURATION (HOURS)††	ESTIMATED DOSE RATE (MREM/HR) (mSv/h)	OCCUPATIONAL DOSE TO INDIVIDUAL (PERSON-MREM) (PERSON-mSv)	ESTIMATED TOTAL DOSE FOR TASK (PERSON-MREM) (PERSON-mSv)
VACUUM CELLS OF MPC	2	1.5	5 (0.05)	7.5 (0.075)	15 (0.15)
REMOVE HI-STAR FROM SPENT FUEL POOL	3	1.2	5 (0.05)	6 (0.06)	18 (0.18)
LOWER WATER LEVEL IN MPC	1	0.2	5 (0.05)	1 (0.01)	1 (0.01)
PUMP REMAINING WATER IN MPC TO SPENT FUEL POOL	1	2	0 (0)	0 (0)	0 (0)
REMOVE MPC FROM HI-STAR	2	1	0 (0)	0 (0)	0 (0)
DECONTAMINATE MPC AND HI-STAR	3	2	0 (0)	0 (0)	0 (0)
TOTAL					934.2 (9.342)

† Indicates number of workers in direct or close contact with HI-STAR 100.
†† Indicates actual duration of work in direct or close contact with HI-STAR 100.

Table 10.3.3
ESTIMATED EXPOSURES FOR HI-STAR 100 SYSTEM SURVEILLANCE AND MAINTENANCE
(40,000MWD/MTU, 5-YEAR COOLED FUEL)

ACTIVITY	ESTIMATED PERSONNEL	ESTIMATED HOURS PER YEAR	ESTIMATED DOSE RATE (MREM/HR) (mSv/h)	OCCUPATIONAL DOSE TO INDIVIDUAL (PERSON-MREM) (PERSON-mSv)	ESTIMATED TOTAL DOSE FOR TASK (PERSON-MREM) (PERSON-mSv)
SECURITY SURVEILLANCE	1	30	4 (0.04)	120 (1.2)	120 (1.2)
ANNUAL MAINTENANCE	2	15	50 (0.5)	750 (7.5)	1500 (15)

10.4 ESTIMATED COLLECTIVE DOSE ASSESSMENT

10.4.1 Controlled Area Boundary Dose for Normal Operations

10CFR72.104 [10.0.1] limits the annual dose to any real individual at the controlled area boundary to a maximum of 25 mrem (0.25 mSv) to the whole body, 75 mrem (0.75 mSv) to the thyroid, and 25 mrem (0.25 mSv) for any other *critical* organ. This includes contributions from all uranium fuel cycle operations in the region.

It is not feasible to predict bounding controlled area boundary dose rates on a generic basis since radiation from plant and other sources; the location and the layout of an ISFSI; and the number and configuration of casks are necessarily site-specific. In order to compare the performance of the HI-STAR 100 System with the regulatory requirements, sample ISFSI arrays were analyzed in Chapter 5. These represent a full array of design basis fuel assemblies. Users are required to perform a site specific dose analysis for their particular situation in accordance with 10CFR72.212 [10.0.1]. The analysis must account for the ISFSI (size, configuration, fuel assembly specifics) and any other radiation from uranium fuel cycle operations within the region.

Table 5.1.7 presents dose rates at various distances from sample ISFSI arrays for the design basis burnup and cooling time which results in the highest off-site dose for the combination of maximum burnup and minimum cooling times analyzed in Chapter 5. 10CFR72.106 [10.0.1] specifies that the minimum distance from the ISFSI to the controlled area boundary is 100 meters. Therefore this was the minimum distance analyzed in Chapter 5. As a summary of Chapter 5, Table 10.4.1 presents the annual dose results for a single cask at 100, 251, and 300 meters and a 2x5 array of HI-STAR 100 systems at 400 meters. These annual doses are conservatively based on a full array of design basis fuel with a burnup of 40,000 MWD/MTU and 5-year cooling.. In addition, 100% occupancy (8760 hours) is conservatively assumed. In the calculation of the annual dose, a cask-to-cask pitch of 12 feet was assumed and the casks were positioned on an infinite slab of concrete to account for earth-shine effects. These results indicate that the calculated annual dose is less than the regulatory limit of 25 mrem/year (0.25 mSv/yr) at a distance of 300 meters for a single cask and at 400 meters for a 2x5 array of HI-STAR 100 Systems containing design basis fuel. The calculated annual dose is 25 mrem (0.25 mSv) at 251 meters. These results are presented only as an illustration to demonstrate that the HI-STAR 100 System is in compliance with 10CFR72.104[10.0.1]. Neither the distances nor the array configurations become part of the Technical Specifications. Rather, users are required to perform a site specific analyses to demonstrate compliance with 10CFR72.104[10.0.1] contributors and 10CFR20[10.1.1].

Section 7.1 provides a discussion as to how the Holtec MPC design, welding, testing, and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible. Therefore, there is no additional dose contribution due to leakage from the welded MPC. The site licensee is required to perform a site-specific dose evaluation of all dose contributors as part of the ISFSI design as dictated in Chapter 12. This evaluation will account for the location of the controlled area boundary and the effects of the radiation from uranium fuel cycle operations within the region.

10.4.2 Controlled Area Boundary Dose for Accident Conditions

10CFR72.106 [10.0.1] specifies *the maximum doses allowed to any individual at the controlled area boundary from any design basis accident (see Subsection 10.1.2)*. In addition, it is specified that the minimum distance from the ISFSI to the controlled area boundary be at least 100 meters.

Chapter 11 presents the results of the evaluations performed to demonstrate that the HI-STAR 100 System can withstand the effects of all credible accident conditions and natural phenomena without the corresponding radiation doses exceeding the requirements of 10CFR72.106 [10.0.1]. The accident events addressed in Chapter 11 include: HI-STAR 100 handling accident, tip-over, fire, tornado, flood, earthquake, 100 percent fuel rod rupture, explosion, lightning, burial under debris, and extreme environmental temperature. The worst-case shielding consequence of the accidents evaluated in Chapter 11 assumes that as a result of a fire, the neutron shield is completely destroyed and replaced by a void. The neutron shield is assumed to be completely lost, whereas some portion of the neutron shield would be expected to remain, as the neutron shield material is fire retardant. The shielding analysis of the HI-STAR 100 System with complete loss of the neutron shield is discussed in Section 5.1.2. The results in that section, show that the resultant dose rate at the 100-meter controlled area boundary would be less than 5 mrem/hr (0.05 mSv/hr) for a single HI-STAR 100 during the accident condition. At this level, it would take more than 1000 hours (41days) for the dose at the controlled area boundary to reach 5 rem (0.05 mSv). This length of time greatly exceeds the time necessary to implement and complete the corrective actions outlined in Chapter11. Therefore, the dose requirement of 10CFR72.106 [10.0.1] is satisfied.

Table 10.4.1
ANNUAL DOSE FOR ARRAYS OF HI-STAR 100
WITH DESIGN BASIS ZIRCALOY CLAD FUEL
40,000 MWD/MTU AND 5-YEAR COOLING

Array Configuration	**1 Cask**	**1 Cask**	**1 Cask**	**2x5 Array**
Annual Dose (mrem/year)† (mSv/year)	345.00 (3.45)	25.00 (0.25)	13.55 (0.1355)	23.06 (0.2306)
Distance to Controlled Area Boundary (meters)††, †††	100	251	300	400

† 100% occupancy is assumed.
†† Dose location is at the center of the long side of the array.
††† Actual controlled area boundary dose rates will be lower because the maximum permissible burnup for 5-year cooling as specified in Figure 2.1.6 and Table 2.1.11 is lower than the burnup analyzed for the design basis fuel used in this table.

10.5 REGULATORY COMPLIANCE

The HI-STAR 100 System provides radiation shielding and confinement features that are sufficient to meet the requirements of 10CFR72.104 and 10CFR72.106 [10.0.1].

Occupational radiation exposures satisfy the limits of 10CFR20 [10.1.1] and meet the objective of maintaining exposures ALARA.

The design of the HI-STAR 100 System is in compliance with 10CFR72 [10.0.1] and applicable design and acceptance criteria have been satisfied. The radiation protection system design provides reasonable assurance that the HI-STAR 100 System will allow safe storage of spent fuel.

10.6 REFERENCES

[10.0.1] *U.S. Code of Federal Regulations,* "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Part 72, "*Energy*."

[10.0.2] U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems", NUREG-1536, Final Report, January 1997.

[10.0.3] Holtec International Final Safety Analysis Report for the HI-STORM 100 Cask System, Revision 12

[10.0.4] Holtec International Final Safety Analysis Report for the HI-STORM FW Cask System, Revision 3

[10.1.1] *U.S. Code of Federal Regulations,* "Standards for protection Against Radiation," Part 20, "*Energy*."

[10.1.2] U.S. Nuclear Regulatory Commission "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power at Nuclear Power Stations will be As Low As Reasonably Achievable", Regulatory Guide 8.8, June 1978.

[10.1.3] U.S. Nuclear Regulatory Commission, "Operating Philosophy for Maintaining Occupational Radiation Exposures As low As is Reasonably Achievable", Regulatory Guide 8.10, Revision 1-R, May 1997.

CHAPTER 11: ACCIDENT ANALYSIS†

This chapter presents the evaluation of the HI-STAR 100 System for the effects of off-normal and postulated accident conditions. The design basis off-normal and postulated accident events, including those resulting from mechanistic and non-mechanistic causes as well as those caused by natural phenomena, are identified in Subsections 2.2.2 and 2.2.3. For each postulated event, the event cause, means of detection, consequences, and corrective action are discussed and evaluated. As applicable, the evaluation of consequences includes structural, thermal, shielding, criticality, confinement, and radiation protection evaluations for the effects of each design event.

The structural, thermal, shielding, criticality, and confinement features and performance of the HI-STAR 100 System are discussed in Chapters 3, 4, 5, 6, and 7, respectively. The evaluations provided in this chapter are based on the design features and evaluations described therein.

The safety analyses summarized in this chapter demonstrate acceptable margins to the allowable limits under all design basis loading conditions and operational modes. Minor changes to the design parameters that inevitably occur during the product's life cycle which are treated within the purview of 10CFR72.48 and are ascertained to have an insignificant effect on the computed safety factors may not prompt a formal reanalysis and revision of the results and associated data in the tables of this chapter unless the cumulative effect of all such un-quantified changes on the reduction of any of the computed safety margins cannot be deemed to be insignificant. For purposes of this determination, an insignificant loss of safety margin with reference to an acceptance criterion is defined as the estimated reduction that is no more than one order of magnitude below the available margin reported in the FSAR. To ensure rigorous configuration control, the information in the Licensing drawings in Section 1.5 should be treated as the authoritative source for numerical analysis at all times. Reliance on the input data and associated results in this chapter for additional mathematical computations may not be appropriate as they serve the sole purpose of establishing safety compliance in accordance with the acceptance criteria set down in Chapter 2 and in this chapter.

Some of the new information in this chapter is directly extracted from previously NRC approved Holtec dockets; this information is shown in italics. In Chapter 4, this information was extracted from References [11.0.1] through [11.0.3]. All changes in this revision are marked with revision bars.

† Wherever multiple units are shown, the US units are the governing value, and the SI units are shown for information only.

11.1 OFF-NORMAL OPERATIONS

During normal storage operations of the HI-STAR 100 System it is possible that an off-normal situation could occur. Off-normal operations, as defined in accordance with ANSI/ANS-57.9, are those conditions which, although not occurring regularly, are expected to occur no more than once a year. In this section, design events pertaining to off-normal operation for expected operational occurrences are considered.

The following off-normal operation events have been considered in the design of the HI-STAR 100, as listed in Subsection 2.2.2:

Off-Normal Pressures
Off-Normal Environmental Temperatures
Leakage of One MPC Seal Weld
Malfunction of Forced Helium Dehydrator (FHD)

For each event, the postulated cause of the event, detection of the event, analysis of the event effects and consequences, corrective actions, and radiological impact from the event are presented.

The results of the evaluations performed herein demonstrate that the HI-STAR 100 System can withstand the effects of off-normal events without affecting the design function, and are in compliance with the applicable acceptance criteria. The section demonstrates that no instruments or controls are required to remain operational under all credible off-normal conditions. The following sections present the evaluation of the HI-STAR 100 System for the design basis off-normal conditions which demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses satisfy the requirements of 10CFR72.106(b) and 10CFR20.

The structural load combinations evaluated for off-normal conditions are defined in Table 2.2.14. The load combinations include both normal and off-normal loads. The off-normal load combination evaluations are discussed in Subsection 11.1.4.

11.1.1 Off-Normal Pressures

There are three pressure regions in the HI-STAR 100 System and they are the MPC internal, the MPC external/overpack internal, and the overpack external pressure regions. Off-normal pressure at these three locations is evaluated at the point at which they act. The MPC internal pressure affects the MPC internal cavity. The MPC external/overpack internal pressure affects the MPC exterior and the overpack internal cavity. The overpack external pressure affects the exterior of the overpack.

11.1.1.1 Postulated Cause of Off-Normal Pressure

The off-normal pressure for the MPC internal cavity is a function of the initial helium fill pressure and the temperature obtained with maximum decay heat load design basis fuel. The maximum off-normal environmental temperature is 100°F (38°C) with full solar insolation. The MPC internal pressure is further increased by the conservative assumption that 10% of the fuel rods rupture, 100% of the fill gas, and fission gases (30%) per NUREG-1536 are released to the cavity.

There is no cause or postulated cause for an off-normal MPC external/overpack internal pressure. There is no cause or postulated cause for off-normal overpack external pressure. Therefore, no off-normal overpack external pressure or off-normal MPC external/overpack internal pressure is evaluated.

11.1.1.2 Detection of Off-Normal Pressure

The HI-STAR 100 System is designed to withstand the MPC off-normal pressure without any effects on its ability to meet its safety requirements. There is no requirement for detection of off-normal pressure in the MPC.

11.1.1.3 Analysis of Effects and Consequences of Off-Normal Pressure

For vertically-oriented systems, Chapter 4 calculates the MPC internal pressure with an ambient temperature of 80°F (27°C), 10% fuel rods ruptured, full insolation, and maximum decay heat and reports the maximum value in Table 4.4.15 at an average calculated MPC cavity temperature of 499.2°K. Using this pressure, the off-normal temperature of 100°F (38°C) (ΔT of 20°F or 11.1°K), and the ideal gas law, the off-normal resultant pressure is calculated to be below the normal condition MPC internal design pressure, as follows:

$$\frac{P_1}{P_2} = \frac{T_1}{T_2}$$

$$P_2 = \frac{P_1 T_2}{T_1}$$

$$P_2 = \frac{(60.2\,psig + 14.7)(499.2°K + 11.1°K)}{499.2°K}$$

$$P_2 = 76.6\,psia \;\; or \;\; 61.9\,psig \; or \;\; 427\,kPa$$

For horizontally-oriented systems, Chapter 3 the HI-STAR 100 transport SAR considers 100°F (38°C) as the normal condition ambient temperature, so there is no need to consider any increase in ambient temperature for this off-normal pressure event. As such, Table 3.4.15 of the HI-STAR 100 transport SAR reports maximum normal operating pressures for an ambient temperature of 100°F (38°C) and no fuel rod ruptures as 87.7 psig (605 kPa) and, with 3% fuel rods ruptured, 89.3 psig (616kPa). For the 10% rod ruptured condition of the off-normal condition, the pressure can be calculated as:

$$P_2 = (89.3\,psig - 87.7\,psig) \times \frac{10\%}{3\%} + 87.7\,psig = 93.0\,psig \; or \; 641\,kPa$$

The normal condition MPC internal pressure of 100 psig (689 kPa) (Table 2.2.1) has been established to bound the off-normal condition. Therefore, no additional analysis is required. The normal condition design pressure, which is equal to the off-normal design pressure, is analyzed in Chapter 3 for Load Case E1. The results in Chapter 3 show that the stress values are below the normal condition allowables.

Structural

The structural evaluation of the MPC enclosure vessel for off-normal design internal pressure conditions is equivalent to the evaluation at normal design internal pressures, since the normal design pressure was set at a value which would encompass the off-normal condition. Therefore, the resulting stresses from the off-normal design condition are equivalent to that of the normal design condition and are well within the allowable stress limits, as discussed in Section 3.4.

Thermal

The MPC internal pressure for off-normal conditions is calculated as presented above. As can be seen from the value calculated above, the 100 psig (689 kPa) design basis internal pressure for off-normal conditions used in the structural evaluation bounds the calculated value.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the off-normal pressure does not affect the safe operation of the HI-STAR 100 System.

11.1.1.4 Corrective Action for Off-Normal Pressure

The HI-STAR 100 System is designed to withstand the off-normal pressure without any effects on its ability to maintain safe storage conditions. There is no corrective action requirement for off-normal pressure.

11.1.1.5 Radiological Impact of Off-Normal Pressure

The event of off-normal pressure has no radiological impact because the confinement barrier and shielding integrity are not affected.

11.1.2 Off-Normal Environmental Temperatures

The HI-STAR 100 System is designed for use at any site in the contiguous United States. Off-normal environmental temperature extremes of -40 and 100 degrees F (-40 and 38 degrees C) have been conservatively selected to bound off-normal temperatures at these sites. The off-normal temperature range affects the entire HI-STAR 100 System and must be evaluated against the allowable component design temperatures. This off-normal event is of a short duration and therefore, the resultant temperatures are evaluated against the off-normal condition temperature limits as listed in Table 2.2.3.

11.1.2.1 Postulated Cause of Off-Normal Environmental Temperatures

The off-normal environmental temperature is postulated as a constant ambient temperature caused by extreme weather conditions. To determine the effects of the off-normal temperatures, it is conservatively assumed that these temperatures persist for a sufficient duration to allow the HI-STAR 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STAR 100 System with its corresponding large thermal inertia and the limited duration for the off-normal temperatures, this assumption is conservative.

11.1.2.2 Detection of Off-Normal Environmental Temperatures

The HI-STAR 100 System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There is no requirement for detection of off-normal environmental temperatures.

11.1.2.3 Analysis of Effects and Consequences of Off-Normal Environmental Temperatures

The off-normal event considering an environmental temperature of 100°F (38°C) for a duration sufficient to reach thermal equilibrium is evaluated with respect to design temperatures listed in Table 2.2.3. The evaluation is performed with design basis fuel with the maximum decay heat and the most restrictive thermal resistance. The 100°F (38°C) environmental temperature is applied with full solar insolation.

For vertically-oriented HI-STAR 100 Systems, the maximum temperatures for components that have temperatures close to their design basis temperatures are listed in Tables 4.4.10 and 4.4.11. These temperatures are conservatively calculated at an environmental temperature of 80°F (27°C). The maximum off-normal environmental temperature is 100°F (38°C), which is an increase of 20°F (11°C). The bounding off-normal component temperatures are calculated by adding 20°F (11°C) to the maximum normal temperatures from the highest component temperature from either the MPC-68 or the MPC-24 (whichever bounds). Table 11.1.1 lists the maximum off-normal temperatures. As illustrated by the table, all the maximum off-normal temperatures are well below the off-normal condition design basis temperatures. Under these conditions, the HI-STAR 100 System maximum off-normal temperatures meet the design requirements specified in Table 2.2.3.

For horizontally-oriented systems, Chapter 3 the HI-STAR 100 transport SAR considers 100°F (38°C) as the normal condition ambient temperature. As described in Section 4.4.2, however, this transport condition is bounding for normal storage conditions. Therefore, it is necessary to increase the horizontal transport temperatures by 20°F (11°C) to account for off-normal conditions of storage. Table 11.1.1 lists the temperatures and all are well below off-normal condition design basis temperatures.

In addition, the off-normal environmental temperature generates a pressure which is evaluated in Section 11.1.1. The off-normal MPC cavity pressure is less than the design basis normal/off-normal pressures listed in Table 2.2.1.

The off-normal event considering an environmental temperature of -40°F (-40°C), no decay heat, and no solar insolation for a duration sufficient to reach thermal equilibrium is evaluated with respect to material design temperatures. The HI-STAR 100 System is conservatively assumed to reach -40°F (-40°C) throughout the structure. All structural analysis is performed at the material design basis temperature, which is set higher than the component would experience with the design basis heat load under normal conditions. Assuming the HI-STAR 100 System is -40°F (-40°C) would only serve to increase the safety margins as the material strength increases with decreasing temperatures. Subsection 3.1.2.3 details the structural analysis performed to evaluate brittle fracture at the lowest service temperature. Subsection 3.4.5 provides a structural evaluation of the effects of an environmental temperature of -40°F (-40°C) and demonstrates that there is no reduction in the performance of the HI-STAR 100 System. Based on this evaluation, it is concluded that the off-normal environmental temperatures do not affect the safe operation of the HI-STAR 100 System.

Structural

The effect on the MPC for the maximum off-normal temperature condition is an increase in the internal pressure. As shown in Section 11.1.1.3, the resultant pressure is well below the normal/off-normal design pressure of 100 psig (689 kPa) used in the structural analysis. The effect of the minimum off-normal temperature conditions results in an evaluation of the potential for brittle fracture which is discussed in Section 3.1.2.3.

Thermal

The resulting off-normal system and fuel assembly cladding temperatures for the hot conditions are provided in Table 11.1.1. As can be seen from these tables, all temperatures for off-normal conditions are within the allowable values described in Table 2.2.3.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal environmental temperatures do not affect the safe operation of the HI-STAR 100 System.

11.1.2.4 Corrective Action for Off-Normal Environmental Temperatures

The HI-STAR 100 System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There are no corrective actions required for off-normal environmental temperatures.

11.1.2.5 Radiological Impact of Off-Normal Environmental Temperatures

Off-normal environmental temperatures have no radiological impact as the confinement barrier and shielding integrity are not affected.

11.1.3 Leakage of One Seal

The HI-STAR 100 System has multiple boundaries to contain radioactive fission products within the confinement boundary and the helium atmosphere within the helium retention boundary (overpack internal cavity). The radioactive material confinement boundary is defined by the MPC shell, baseplate, MPC lid, and vent and drain cover plates. The closure ring provides a redundant welded closure to prevent the release of radioactive material from the MPC cavity. Confinement boundary welds, including the MPC lid-to-shell weld, are inspected by radiography or ultrasonic examination except for field welds on the closure ring and vent/drain port cover plates. The closure ring and vent/drain port cover plates are examined by the liquid penetrant method on the root (for multi-pass welds) and final pass. The welds on the MPC lid, vent and drain port covers are leakage tested. The MPC is also hydrostatically tested. There are no seals present in the design of the MPC confinement boundary.

Section 7.1 provides the narrative that demonstrates that the MPC design, welding, testing and inspection meet the requirements such that leakage from the helium retention boundary is considered non-credible.

11.1.4 Off-Normal Load Combinations

Structural load combinations for off-normal conditions are provided in Table 2.2.14. The load combinations include normal loads with the off-normal loads. The load combination results are shown in Section 3.4 to meet all allowable values.

11.1.5 Malfunction of FHD System

The FHD system is a forced helium circulation device used to effectuate moisture removal from loaded MPCs. For circulating helium, the FHD system is equipped with active components requiring external power for normal operation.

11.1.5.1 Postulated Cause of FHD Malfunction

Likely causes of FHD malfunction are (i) a loss of external power to the FHD System and (ii) an active component trip. In both cases a stoppage of forced helium circulation occurs. Such a circulation stoppage does not result in any helium leakage from the MPC or the FHD itself.

11.1.5.2 Detection of FHD Malfunction

The FHD System is monitored during its operation. An FHD malfunction is detected by operator response to control panel visual displays and alarms.

11.1.5.3 Analysis of Effects and Consequences of FHD Malfunction

Structural

The FHD System is required to be equipped with safety relief devices[1] to prevent the MPC structural boundary pressures from exceeding the design limits. Consequently there is no adverse effect.

1 The relief pressure is below the off-normal design pressure (Table 2.2.1) to prevent MPC overpressure and above MNOP to enable MPC pressurization for adequate heat transfer.

Thermal

Malfunction of the FHD System is categorized as an off-normal condition, for which the applicable peak cladding temperature limit is 1058ºF or 570ºC (Table 2.2.3). The FHD System malfunction event is evaluated assuming the following bounding conditions:

1) Steady state maximum temperatures have been reached
2) Design basis heat load
3) MPCs backfilled with the minimum helium pressure required by the Technical Specifications

It is noted that operator action may be required to raise the helium regulator set point to ensure that condition 3 above is satisfied. These conditions are the same as for the normal storage in a vertically-oriented system, discussed in Section 4.4. The steady state results are provided in Tables 4.4.10 and 4.4.11. The results demonstrate that the peak fuel cladding temperatures remain below the limit in the event of a prolonged unavailability of the FHD system.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation above, the structural boundary pressures cannot exceed the design limits.

Radiation Protection

As there is no adverse effect on the shielding or confinement functions, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the FHD malfunction does not affect the safe operation of the HI-STAR 100 System.

11.1.5.4 Corrective Action for FHD Malfunction

The HI-STAR 100 System is designed to withstand the FHD malfunction without an adverse effect on its safety functions. Consequently no corrective action is required.

11.1.5.5 Radiological Impact of FHD Malfunction

The event has no radiological impact because the confinement barrier and shielding integrity are not affected.

Table 11.1.1

MAXIMUM TEMPERATURES CAUSED BY OFF-NORMAL ENVIRONMENTAL TEMPERATURES (°F [°C])

Temperature Location	**Normal**	**Calculated Off-Normal**	**Off-Normal Design Basis Limits**
Vertically-Oriented Systems			
Fuel cladding	741† [394]	761 [405]	1058 [570]
MPC basket	725† [385]	745 [396]	950 [510]
MPC outer shell surface	332†† [167]	352 [178]	775 [410]
MPC/overpack helium gap outer surface	292†,†† [144]	312 [156]	500 [260]
Neutron shield inner surface	274†† [134]	294 [146]	300 [149]
Overpack shell outside surface	229†† [109]	249 [121]	1000 [535]
Horizontally-Oriented Systems			
Fuel cladding	713[1] [378]	733 [389]	1058 [570]
MPC basket	697[1] [369]	717 [381]	950 [510]
MPC outer shell surface	315[2] [157]	335 [168]	775 [410]
MPC/overpack helium gap outer surface	291[2] [144]	311 [155]	500 [260]
Neutron shield inner surface	271[2] [133]	291 [144]	300 [149]
Overpack shell outside surface	222[2] [106]	242 [117]	1000 [535]

† MPC-68 normal storage maximum temperatures from Table 4.4.11.
†† MPC-24 normal storage maximum temperatures from Table 4.4.10.
1 BWR MPC temperature from HI-STAR 100 SAR Table 3.4.11.
2 PWR MPC temperature from HI-STAR 100 SAR Table 3.4.10.

11.2 ACCIDENTS

Accidents, in accordance with ANSI/ANS-57.9, are either infrequent events that could reasonably be expected to occur during the lifetime of the HI-STAR 100 System or events postulated because their consequences may affect the public health and safety. Section 2.2.3 defines the design basis accidents considered. By analyzing for these design basis events, safety margins inherently provided in the HI-STAR 100 System design can be quantified.

The results of the evaluations performed herein demonstrate that the HI-STAR 100 System can withstand the effects of all credible accident conditions and natural phenomena without affecting safety function, and are in compliance with the acceptable criteria. The section demonstrates that no instruments or controls are required to remain operational under all credible accident conditions. The following sections present the evaluation of the design basis postulated accident conditions and natural phenomena which demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses satisfy the requirements of 10CFR72.106(b) and 10CFR20.

The structural load combinations evaluated for postulated accident conditions are defined in Table 2.2.14. The load combinations include normal loads with the accident loads. The accident load combination evaluations are provided in Section 3.4.

11.2.1 Handling Accident

During the operation of the HI-STAR 100 System, the loaded overpack is transported to the ISFSI in the vertical or horizontal position. The loaded overpack is typically transported by a heavy-haul vehicle which cradles the overpack horizontally or holds the overpack vertically. A vertical or horizontal drop of the loaded HI-STAR cask is not a credible accident as the loaded HI-STAR cask shall be lifted and handled by devices designed in accordance with the criteria specified in Paragraph 2.2.3.1 to prevent uncontrolled lowering. Therefore, postulating an uncontrolled lowering of a HI-STAR cask in the realm of Part 72 operations is non-credible.

11.2.2 Tip-Over

11.2.2.1 Cause of Tip-Over

The analysis of the HI-STAR 100 System has shown that a vertically-oriented cask does not tip over as a result of the accidents (i.e., tornado missiles, flood water velocity, and seismic activity) analyzed in this section. The tip-over accident for a vertically-oriented cask is stipulated as a non-mechanistic accident.

As stated in Subparagraph 2.2.3.2, tip-over is not a credible event for a horizontally-oriented cask. The remainder of this discussion of a tip-over event therefore only applies to a vertically-oriented cask.

11.2.2.2 Tip-Over Analysis

The tip-over accident analysis evaluates the effects of the loaded overpack tipping-over onto a reinforced concrete pad. The tip-over analysis is provided in Chapter 3. The analysis shows that the HI-STAR 100 System meets all structural requirements and there is no adverse effect on the confinement, thermal, or subcriticality performance of the cask. However, the tip-over could cause some damage to the overpack outer enclosure shell and neutron shield in the area of impact.

Structural

Appendix 3.A calculates the maximum deceleration of the HI-STAR 100 System as a result of a non-mechanistic tip-over. For tip-over analysis of the HI-STAR 100 System onto the ISFSI pad, the analysis presented in Appendix 3.A demonstrates that the deceleration of the MPC remains below 60g's. The structural analyses of the MPC and overpack under a 60g radial load are presented Section 3.4 and it is demonstrated therein that the allowable stresses are within allowable limits.

Thermal

As the structural analysis demonstrates that there is no change in the MPC or overpack except for localized neutron shield damage, there is a negligible effect on the thermal performance of the system as a result of this event.

Shielding

Localized damage of the radial neutron shield is to be expected as a result of the tip-over. The damage will be limited to the impacted area.

Criticality

As the structural analysis demonstrates that there is no change in the MPC or overpack, there is a negligible effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is a very localized reduction in shielding and no effect on the confinement capabilities as discussed above, there is a negligible effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the non-mechanistic tip-over of the HI-STAR 100 System does not affect its safe operation.

11.2.2.3 Tip-Over Dose Calculations

The tip-over accident could cause localized damage to the neutron shield and outer enclosure shell where the neutron shield impacts the ISFSI pad. The gamma shielding will not be affected. The overpack surface dose rate in the affected area could increase due to damage of the neutron shield. However, there should be no noticeable increase in the ISFSI site or controlled area boundary dose rate, because the affected areas will likely be small. Once the overpack is uprighted, some local dose increase could occur. The cask post-accident shielding analysis in Chapter 5 assumes complete loss of the neutron shield and bounds the dose rates anticipated for the tip-over accident. The analysis of the tip-over accident has shown that the MPC confinement barrier will not be compromised and, therefore, there will be no release of radioactivity.

11.2.2.4 Tip-Over Accident Corrective Action

The handling accident corrective action procedure outlined in Subsection 11.2.1.4 is applicable for the recovery of the tip-over accident.

11.2.3 Fire

11.2.3.1 Cause of Fire

Although the probability of a fire accident affecting a HI-STAR 100 System during storage operations is low due to the lack of combustible materials at the ISFSI, a fire resulting from an on-site transporter fuel tank contents is postulated and analyzed. The analysis shows that the HI-STAR 100 System continues to perform its structural, confinement, and subcriticality functions.

11.2.3.2 Fire Analysis

The thermal environment to which the HI-STAR 100 System would be exposed under a hypothetical fire accident is specified to be the same as that required in 10CFR71.73(c)(4). The overpack surfaces are therefore considered to receive an incident thermal radiation and convective heat flux from an ambient 1475°F (800°C) fire condition environment. The duration of fire resulting from an on-site transporter fuel tank spill is calculated as follows:

Volume of Fuel (V) = 50 gallons (6.68 ft^3 or 189 L) (Specified by Subsection 2.2.3.3)

Overpack Baseplate (D_i) = 83-1/4" (6.9375 ft or 211 cm) (Overpack Drawing, Section 1.5)

Fuel Spill Ring Width (L) = 1 meter (IAEA Specification [11.2.6])

$$\text{Fuel Spill Diameter}\,(D_o) = 83\ 1/4" + 2\text{m} \times \frac{1"}{0.0254\text{m}}$$

= 161.99” (13.4991 ft or 411 cm)

$$\text{Fuel Spill Area}\,(A) = \frac{\pi}{4}(D_o^2 - D_i^2)$$

= 105.3 ft^2 (9.78 m^2)

$$\text{Spill Depth}\,(d) = \frac{V}{A} = \frac{6.68}{105.3}$$

= 0.0634 ft (0.761” or 1.93 cm)

Fuel Consumption Rate (R) = 0.15 inch/min (0.38 cm/min) [11.2.7]

$$\text{Fire Duration} = \frac{d}{R} = \frac{0.761}{0.15}$$

= 5.075 min (305 seconds)

It is noted that this duration is calculated for a vertically-oriented cask. This bounds the duration for a horizontally-oriented cask as follows. This design-basis fire is defined using the guidance of 10CFR71.73, which is endorsed by NUREG-1536. As described therein, the fuel puddle must extend horizontally at least 1 m (40 in) beyond any external surface of the cask. For a vertical HI-STAR 100 the external surface of the cask is defined by a 93 3/4” circle. For a horizontal HI-STAR 100 the external surface of the cask is defined by a 93 3/4” x 191 1/8” rectangle. The fuel puddle formed around the much larger rectangular area will also be much larger, and therefore shallower, than that formed around the vertical cask. A shallower puddle results in a shorter fire, as the diameter of the cask is less than its length and so a vertically-oriented cask results in a significantly smaller fuel puddle, which maximizes the puddle depth and corresponding fire duration. As a result the temperature response of the cask is evaluated using the duration for a vertically-oriented cask.

Within this time period, the cask outside surface and its contents will undergo a transient temperature rise due to the heat absorbed from the fire. Full effects of insolation before, during, and after the fire are included in the HI-STAR 100 System transient analysis. During the postulated fire event, the neutron shield material is exposed to high temperatures. Therefore, conservatively, an upper bound material thermal conductivity is assumed during the fire to maximize heat input to the cask. During the post-fire cooldown phase, no credit is taken for conduction through the neutron shield. The temperature history of a number of critical points in the HI-STAR 100 System transient fire analysis are tracked during the fire and the subsequent relaxation of temperature profiles during the post-fire cooldown phase. The impact of transient temperature excursions on HI-STAR 100 System materials is assessed in this section. During the fire, a cask surface emissivity specified in 10CFR71.73(b)(4) is applied to maximize radiant heat input. Destruction of the paint covering the external cask surfaces due to exposure to intense heat during fire is a credible possibility. Therefore, a lower emissivity of the exposed carbon steel surface is conservatively applied for post-fire cooldown analysis. This approach provides a conservatively bounding response of the HI-STAR 100 System to the fire accident condition.

Heat input from the fire to the HI-STAR 100 System is from a combination of radiation and convection heat transfer to all overpack exposed surfaces. This can be expressed by the following equation:

$$q_F = h_{fc}\left(T_F - T_S\right) + 0.1714\varepsilon\left[\left(\frac{T_F + 460}{100}\right)^4 - \left(\frac{T_S + 460}{100}\right)^4\right]$$

where:

q_F = surface heat input flux (Btu/ft^2-hr)

T_F = fire condition temperature (1475°F)

T_S = transient surface temperature (°F)

h_{fc} = forced convection heat transfer coefficient [Btu/ft^2-hr-°F]

ε = surface emissivity = 0.9 (per 10CFR71)

The forced convection heat transfer coefficient is calculated to bound the convective heat flux contribution to the exposed cask surfaces due to fire induced air flow. For the case of air flow past a heated cylinder, Jacob [11.2.3] recommends the following correlation for convective heat transfer obtained from experimental data:

$$Nu_{fc} = 0.028\,\mathrm{Re}^{0.8}\left[1 + 0.4\left(\frac{L_{st}}{L_{tot}}\right)^{2.75}\right]$$

where:

L_{tot} = length traversed by flow

L_{st} = length of unheated section

K_f = thermal conductivity of air evaluated at the average film temperature

Re = flow Reynolds Number based on L_{tot}

Nu_{fc} = Nusselt Number ($h_{fc}\ L_{tot}/K_f$)

A consideration of the wide range of temperatures to which the exposed surfaces are subjected during fire and the temperature dependent trend of air properties requires a careful selection of parameters to determine a conservatively large bounding value of the convective heat transfer coefficient. Table 11.2.1 provides a summary of parameter selections with justifications which provide the basis for application of this correlation to determine the forced convection heating of the HI-STAR 100 System during this short-term fire event.

After the fire event, the outside environment temperature is restored to initial ambient conditions and the HI-STAR 100 System transient analysis is continued, to evaluate temperature peaking in the interior during the post-fire cooldown phase. Heat loss from the outside exposed surfaces of the overpack is determined by the following equation:

$$q_S = 0.19\left(T_S - T_A\right)^{4/3} + 0.1714\varepsilon\left[\left(\frac{T_S + 460}{100}\right)^4 - \left(\frac{T_A + 460}{100}\right)^4\right]$$

where:

q_s= surface heat loss flux (Btu/ft^2-hr)

T_S = transient surface temperature (°F)

T_A = ambient temperature (100°F)

ε = surface emissivity of exposed carbon steel surface

The FLUENT thermal analysis model was used to perform the fire condition transient analysis. Based on this analysis, the maximum temperature attained in different portions of the cask during the fire followed by a post-fire cooldown are summarized in Table 11.2.2. From the results, it is apparent that due to the large bulk mass and long radial path lengths for flow of heat, the MPC basket centerline temperatures are relatively unaffected by this short duration fire event. However, the overpack enclosure shell and neutron shield material in its immediate vicinity experience a significant temperature increase. The short-duration temperature rise experienced by the periphery of the neutron shield may result in partial loss of its ability to shield neutrons. The neutron shields' inner surface peak transient temperature at the hottest spatial location (314°F or 157°C) is slightly higher than the 300°F (149°C) long-term temperature limit. This short-term elevated temperature exposure, lasting for a few hours, is not expected to significantly degrade the neutron shield materials shielding function at this location. For conservatism the post-fire shielding evaluation takes no credit for the neutron shield material, replacing it with void as described in Subsection 5.1.2. A pressure relief system is provided on the overpack outer enclosure shell to prevent any overpressurization in the neutron shield region during the fire event. Figures 11.2.1 through 11.2.3 plot the transient temperature-time history of HI-STAR 100 components identified as significant for fire accident performance evaluation. Figure 11.2.4 provides an axial temperature plot of the hottest rod in the post-fire cooldown.

Increased pressure of the MPC due to the temperature rise is also considered. From the maximum temperature rise of the MPC during the post-fire cooldown phase, maximum average MPC cavity temperatures are calculated by adding this temperature increment to the initial condition (before start of fire) MPC cavity average temperature for each MPC and applying the ideal gas law. The initial condition MPC cavity average temperatures and pressures have been determined by analytical methods described in Chapter 4 for a vertical cask and in Chapter 3 of the HI-STAR 100 transport SAR for a horizontal cask. Maximum fire accident pressures in the MPC cavity based on a conservatively bounding 216°F (120°C) MPC cavity temperature rise are reported in Table 11.2.3.

As can be seen by Table 11.2.3, the pressure does not exceed the accident condition design basis pressure listed in Table 2.2.1.

To ensure the fuel assemblies can be retrieved by normal means and the fuel arrangement remains subcritical, the MPC fuel basket is shown to be unconstrained for thermal expansion. Table 11.2.5 provides the HI-STAR 100 component temperatures in the post-fire cooldown phase. Using these temperatures, Subparagraph 3.4.4.2.2 demonstrates that the thermal expansion of the MPC fuel basket is unconstrained.

Structural

As discussed above, there are no structural consequences as a result of the fire accident condition.

Thermal

As discussed above, the MPC internal pressure, based on a conservatively bounding fire condition temperature rise, remains below accident condition design pressure. As shown in Table 11.2.2, the peak fuel cladding and material temperatures are well below short-term accident condition allowable temperatures of Table 2.2.3.

Shielding

The assumed complete loss of all the radial neutron shield in the shielding analysis results in an increase in the radiation dose rates at locations adjacent to the neutron shield. The shielding analysis results presented in Section 5.1.2 demonstrate that the requirements of 10CFR72.106 are not exceeded.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

There is no degradation in confinement capabilities of the MPC, as discussed above. There are increases in the dose rates adjacent to the neutron shield. The dose rate at 1 meter from the neutron shield after the neutron shield is replaced by a void is calculated to be less than 500 mrem/hr (Table 5.1.9). Immediately after the fire accident a radiological inspection of the HI-STAR overpack will be performed and temporary shielding installed to limit the exposure to the public. Based on a minimum distance to the controlled area boundary of 100 meters, the dose rate at the controlled area boundary will be less than 5 mrem/hr. Therefore, it is evident that the requirements of 10CFR72.106 (5 Rem) will not be exceeded.

11.2.3.3 Fire Dose Calculations

The analysis of the fire accident shows that the confinement boundary is not compromised and therefore there is no release of radioactive material. The complete loss of the overpack's radial neutron shield is assumed in the shielding analysis for the post-accident HI-STAR 100 System in Chapter 5. The HI-STAR 100 System following a fire accident meets the dose rate requirements of 10CFR72.106. The seals on the overpack will be exposed to short-term high temperature excursions which remain below the maximum design accident temperature limits listed in Table 2.2.3. However, as no radioactive materials are present in the annulus, the loss of the helium retention boundary will have no radiological impact.

11.2.3.4 Fire Accident Corrective Actions

Upon detection of a fire, the ISFSI operator shall take the appropriate immediate corrective actions necessary to extinguish the fire. Fire fighting personnel should take appropriate radiological precautions as the neutron shielding may be damaged and an increased radiation dose could result.

Following the termination of the fire, a visual and radiological inspection of the overpack shall be performed. Specific attention shall be taken during the inspection of the neutron shield. As appropriate, place temporary shielding around the HI-STAR overpack to reduce local dose rates.

If damage to the neutron shield is limited to a localized area, local repairs can be performed to replace the damaged neutron shield material. If damage to the neutron shield is widespread and/or radiological conditions require, the overpack shall be unloaded in accordance with Chapter 8, prior to repair of the neutron shield.

To verify the continued presence of the helium atmosphere within the overpack cavity, perform the procedure specified in Subsection 11.2.1.4.

Following replacement of the neutron shield material, performance of the shielding effectiveness test per Chapter 9, verification of the appropriate helium atmosphere, and leakage testing of the helium retention boundary seals, the overpack shall be certified to return the overpack to service.

11.2.4 Partial Blockage of MPC Basket Vent Holes

Each MPC basket fuel cell wall has elongated vent holes at the bottom and top. The flow holes in the bottom of the fuel basket in each MPC are located to ensure that the amount of crud listed in Table 2.2.8 does not block the internal helium circulation. Therefore the partial blockage of the MPC basket flow holes is not credible.

11.2.5 Tornado

11.2.5.1 Cause of Tornado

The HI-STAR 100 System will be stored on an unsheltered ISFSI concrete pad and subject to environmental conditions. It is possible that the HI-STAR 100 System may experience the extreme environmental conditions of a tornado.

11.2.5.2 Tornado Analysis

The tornado accident has two effects on the HI-STAR 100 System. The tornado winds or tornado missile attempts to tip-over the loaded overpack with high velocity winds exerting a pressure loading or the potential impact of large tornado missiles striking the overpack. The second effect is tornado missiles propelled by high velocity winds which attempt to penetrate the overpack helium retention boundary and damage the shielding.

Chapter 3 provides the analysis of the pressure loading which attempts to tip-over the overpack and the analysis of the effects of the different types of tornado missiles. These analyses show that the loaded overpack does not tip-over as a result of the tornado winds or tornado missiles. The analyses also show that the overpack helium retention boundary is not compromised and only minor shielding damage will be incurred as a result of the tornado missile. The tornado accident had no adverse consequences on the structural, confinement, thermal, or criticality control capabilities of the HI-STAR 100 System.

Structural

Subsection 3.4.8 describes the analysis of the pressure loading which attempts to tip-over the storage overpack and the analysis of the effects of the different types of tornado missiles. These analyses show that the loaded storage overpack does not tip-over as a result of the tornado winds and/or tornado missiles.

Analyses provided in Subsection 3.4.8 also describes that the tornado missiles do not penetrate the overpack helium retention boundary. The result of the tornado missile impact on the overpack is limited to localized damage of the shielding.

Thermal

There is no effect on the thermal performance of the system as a result of this event.

Shielding

The shielding analysis results presented in Section 5.1.2 demonstrate that the requirements of 10CFR72.106 are not exceeded.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not penetrate the overpack and impact the MPC. There may be increases in the local dose rates adjacent to the impact point of the tornado missile. However, this very localized effect will have no effect on the site boundary dose rate. Therefore, it is evident that the requirements of 10CFR72.106 (5 Rem) will not be exceeded.

11.2.5.3 Tornado Dose Calculations

The tornado winds do not tip-over the loaded overpack, damage the shielding materials or the confinement boundary. There is no affect on the radiation dose as a result of the tornado winds. A tornado missile may cause a very localized reduction in the neutron shielding. However, the damage shall have a negligible effect on the controlled area boundary dose and the effects of the tornado missile damage is bounded by the post-accident dose assessment performed in Chapter 5.

11.2.5.4 Tornado Accident Corrective Action

Following exposure of the HI-STAR 100 System to a tornado, the ISFSI operator shall perform a visual and radiological inspection of the overpack. Damage sustained by the neutron shield shall be repaired in accordance with Subsection 11.2.3.4.

11.2.6 Flood

11.2.6.1 Cause of Flood

The HI-STAR 100 System will be located on an unsheltered ISFSI concrete pad. Therefore, it is possible for the storage area to be flooded. The potential sources for the flood water could be unusually high water from a river or stream, a dam break, a seismic event, or a hurricane.

11.2.6.2 Flood Analysis

The flood accident does not adversely affect the criticality, confinement, shielding, or thermal capabilities of the HI-STAR 100 System. The structural analysis shows that the overpack helium retention boundary, and consequently the MPC confinement boundary maintains full integrity. The criticality analysis for normal fuel loading operations with the cask submerged is more reactive. The flood water acts as a radiation shield and will reduce the radiation doses. The thermal consequences of the flood is an increase in the rejection of the decay heat. Since the flood water temperature will be within the off-normal temperature range specified in Table 2.2.2, the thermal transient associated with the initial contact of the flood water with the overpack exterior surface will be bounded by the off-normal operation conditions.

The flood accident affects the HI-STAR 100 System structural analysis in two ways. First, the flood water velocity acts to apply force and an overturning moment which attempts to cause sliding or tip-over of the loaded overpack. Secondly, the flood water depth applies an external pressure to the overpack. Chapter 3 provides the analysis of both of these conditions. The results of the analysis show that the overpack helium retention boundary is not affected, and that the loaded overpack does not slide or tip over if the flood velocity does not exceed the value stated in Table 2.2.8. The HI-STAR 100 design basis accident external pressure far exceeds any pressure due to an actual flood.

Structural

Subsection 3.4.6 provides the analysis of the flood water applying an overturning moment. The results of the analysis show that the loaded overpack does not tip over if the flood velocity does not exceed the value stated in Table 2.2.8.

The structural evaluation of the overpack for the accident condition external pressure (Table 2.2.1) is presented in Subsection 3.4.6 and the resulting stresses from this event are shown to be well within the allowable values.

Thermal

There is no adverse effect on the thermal performance of the system as a result of this event. The thermal consequences of the flood is an increase in the rejection of the decay heat. Since the flood water temperature will be within the off-normal temperature range specified in Table 2.2.2, the thermal transient associated with the initial contact of the flood water with the overpack exterior surface will be bounded by the off-normal operation conditions. This is due to the higher heat transfer capabilities of water compared to air.

Shielding

There is no effect on the shielding performance of the system as a result of this event. The flood water acts as a radiation shield and will reduce the radiation doses.

Criticality

There is no effect on the criticality control features of the system as a result of this event. The criticality analysis is unaffected because under the flooding condition water does not enter the MPC cavity and therefore the reactivity would be less than the loading condition in the fuel pool which is presented in Section 6.1.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the flood accident does not affect the safe operation of the HI-STAR 100 System.

11.2.6.3 Flood Dose Calculations

Since the flood accident produces no leakage of radioactive material and no reduction in shielding effectiveness, there are no adverse radiological consequences.

11.2.6.4 Flood Accident Corrective Action

As shown in the analysis of the flood accident, the HI-STAR 100 System sustains no damage as a result of the flood. At the completion of the flood, the exterior of the overpack should be inspected, cleaned, and recoated, as necessary, to maintain the proper emissivity.

11.2.7 Earthquake

11.2.7.1 Cause of Earthquake

The HI-STAR 100 System may be employed at any reactor facility or ISFSI. It is possible that during the use of the HI-STAR 100 System, the ISFSI may experience an earthquake.

11.2.7.2 Earthquake Analysis

The earthquake accident analysis evaluates the effects of a seismic event on the loaded HI-STAR 100 System. The objective is to determine the stability limits of the HI-STAR 100 System. Based on a static stability criteria, it is shown in Chapter 3 that the HI-STAR 100 System is qualified to seismic activity less than or equal to the values specified in Table 2.2.8. The analyses in Chapter 3 show that the HI-STAR 100 System will not tip over under the conditions evaluated. The seismic activity has no adverse thermal, criticality, confinement, or shielding consequences.

Structural

The sole structural effect of the earthquake is an inertial loading of less than 1g. This loading is bounded by the handling accident and tip-over analyses presented in Sections 11.2.1 and 11.2.2, which analyzes a deceleration of 60g's and demonstrates that the MPC and overpack allowable stress criteria are met.

Thermal

There is no effect on the thermal performance of the system as a result of this event.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the earthquake does not affect the safe operation of the HI-STAR 100 System.

11.2.7.3 Earthquake Dose Calculations

Structural analysis of the earthquake accident shows that the loaded overpack will not tip over as a result of seismic activity. If the overpack were to tip over, the resultant damage would be equal to that experienced by the tip-over accident analyzed in Subsection 11.2.2. Since the loaded overpack does not tip-over, there is no increase in radiation dose rates or release of radioactivity.

11.2.7.4 Earthquake Accident Corrective Action

Following the earthquake accident, the ISFSI operator shall perform a visual and radiological inspection of the overpacks in storage to determine if any of the overpacks have tipped-over due to the earthquake exceeding the maximum ZPA specified in Chapter 2. In the unlikely event of a tip-over, corrective actions shall be in accordance with Subsection 11.2.1.4.

11.2.8 100% Fuel Rod Rupture

This accident event postulates that all the fuel rod rupture and that the appropriate quantities of fission product gases and fill gas are released from the fuel rods into the MPC cavity.

11.2.8.1 Cause of 100% Fuel Rod Rupture

Through all credible accident conditions, the HI-STAR 100 System maintains the spent nuclear fuel in an inert environment while maintaining the peak fuel cladding temperature below the required short-term temperature limits. There is no credible cause for 100% fuel rod rupture. This accident is postulated to evaluate the MPC confinement barrier for the maximum possible internal pressure.

11.2.8.2 100% Fuel Rod Rupture Analysis

The 100% fuel rod rupture accident has no structural, criticality or shielding consequences. The event does not change the reactivity of the stored fuel, the magnitude of the radiation source which is being shielded, or the shielding capability of the HI-STAR 100 System. The determination of the maximum accident pressure is provided in Chapter 4 for the MPC-24 and MPC-68 and in Chapter 3 of the HI-STAR 100 transport SAR for the MPC-32. The MPC design basis accident internal pressure bounds the pressure developed assuming 100% fuel rod rupture. The structural analysis provided in Chapter 3 evaluates the MPC confinement boundary under the accident condition internal pressure.

As a result of the non-mechanistic 100% fuel rod rupture, the fuel rod fill gas and fission gases are assumed to be released into the MPC cavity. This release causes a dilution of helium by the low thermal conductivity fission gases (Kr, Xe, and Tritium). This dilution of the helium gas and subsequent reduction in the thermal conductivity is bounded by the thermal analysis performed for the vacuum condition during loading operations performed in Chapter 4. Under the vacuum conditions, there is no gas providing a pathway for the thermal conduction of the spent nuclear fuel decay heat. Under the 100% fuel rod rupture condition, the mixture of gases and their resultant lower effective thermal conductivity would provide a thermal conduction pathway. However, no credit is taken for the thermal conductivity of the gas mixture.

From Figure 4.4.19 for the MPC-24 under vacuum conditions, the maximum peak cladding temperature is 691°K and the maximum MPC shell temperature is 384°K. The ΔT between the maximum peak cladding temperature and the maximum MPC shell temperature under vacuum conditions is 307°K or 553°F. The maximum normal condition MPC shell temperature is 332°F (167°C) from Table 4.4.10. Therefore, a bounding peak fuel cladding temperature for the 100% fuel rod rupture may be calculated by adding the ΔT to the maximum normal condition MPC shell temperature. This results in 332°F + 553°F = 885°F (167°C + 307°C = 474°C). This bounding peak fuel cladding temperature is well below the allowable fuel cladding short term temperature limit of 1058°F (570°C).

The most significant thermal consequence of a postulated 100% fuel rod rupture accident is the increase in MPC confinement boundary pressure. As demonstrated in the fire accident transient analysis, the confinement boundary pressure design limit is not exceeded (Table 11.2.3), which includes the 100% fuel and, for the MPC-24 only, PWR BPRA rods rupture.

Structural

The structural evaluation of the MPC for the accident condition internal pressure presented in Section 3.4 demonstrates that the MPC stresses are well within the allowable values.

Thermal

The MPC internal pressure for the 100% fuel rod rupture condition is presented in Table 4.4.15 for the MPC-24 and the MPC-68, and in Chapter 3 of the HI-STAR 100 transport SAR for the MPC-32. Table 11.2.2 provides the MPC internal pressure at fire condition temperatures with 100% fuel rod rupture. As can be seen from the values in both tables, the 200 psig design basis accident condition MPC internal pressure used in the structural evaluation bounds the calculated value.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the non-mechanistic 100% fuel rod rupture accident does not affect the safe operation of the HI-STAR 100 System.

11.2.8.3 100% Fuel Rod Rupture Dose Calculations

The MPC confinement boundary maintains its integrity. There is no effect on the shielding effectiveness, and the magnitude of the radiation source is unchanged. Therefore, there is no release of radioactive material or an increase in radiation dose rates.

11.2.8.4 100% Fuel Rod Rupture Accident Corrective Action

As shown in the analysis of the 100% fuel rod rupture accident, the MPC confinement boundary is not compromised. The HI-STAR 100 System is designed to withstand this accident and continue performing the safe storage of spent nuclear fuel. No corrective actions are required.

11.2.9 Confinement Boundary Leakage

None of the postulated environmental phenomenon or accident conditions identified would cause failure of the confinement boundary. The MPC uses redundant confinement closures to assure that there is no release of radioactive materials. The analyses presented in Chapter 3 and in this chapter demonstrate that the MPC remains intact during all postulated accident conditions. The information contained in Chapter 7 demonstrates that MPC is designed, fabricated, tested and inspected to meet the guidance of ISG-18 such that unacceptable leakage from the Confinement Boundary is noncredible.

11.2.10 Explosion

11.2.10.1 Cause of Explosion

An explosion within the bounds of an ISFSI is improbable since there are no explosive materials stored within the site boundary. An explosion as a result of combustion of the fuel contained in cask transport vehicle is possible. The fuel available for the explosion would be limited by site administrative controls and therefore, any explosion would be limited in size. Any explosion stipulated to occur beyond the site boundary would have a minimal effect on the HI-STAR 100 System. Explosions that are credible for a specific ISFSI would require a site hazards evaluation under the provisions of 72.212 regulations by the ISFSI owner using the methodology set forth in Section 3.4.

11.2.10.2 Explosion Analysis

Any credible explosion accident is bounded by the design basis accident external pressure of 300 psig. The analysis performed in Chapter 3 shows that the HI-STAR 100 System is not adversely affected by the accident condition external pressure.

Structural

The structural evaluations for the overpack accident condition external pressure is presented in Section 3.4 and demonstrates that all stresses are within allowable values.

Thermal

There is no effect on the thermal performance of the system as a result of this event.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the explosion accident does not affect the safe operation of the HI-STAR 100 System.

11.2.10.3 Explosion Dose Calculations

The bounding external pressure load has no effect on the HI-STAR 100 overpack and therefore, no effect on the shielding, criticality, thermal or confinement capabilities of the HI-STAR 100 System.

11.2.10.4 Explosion Accident Corrective Action

The potential overpressure caused by the explosion is bounded by the design basis external pressure. The external pressure from the overpressure is shown not to damage the HI-STAR 100 System. Following an explosion, the ISFSI operator shall perform a visual and radiological inspection of the overpack. If the neutron shield is damaged as a result of explosion generated missiles, the neutron shield material may be replaced and the outer enclosure shell repaired. If damage to the neutron shield is extensive, the damage shall be repaired and retested in accordance with the shielding effectiveness test in Chapter 9.

11.2.11 Lightning

11.2.11.1 Cause of Lightning

The HI-STAR 100 System will be stored on an unsheltered ISFSI concrete pad. There is the potential for lightning to strike the overpack. This analysis evaluates the effects of lightning striking the overpack.

11.2.11.2 Lightning Analysis

The HI-STAR 100 System is a large metallic cask which can be stored in an unsheltered ISFSI. As such, it may be subject to lightning strikes. A lightning strike on the overpack may be visually detected by visible surface discoloration at the point of entry or exit of the current flow. The analysis of the consequence of a lightning strike assumes that the lightning strikes the upper surface of the top flange and proceeds through the inner shell and bottom plate to the ground.

Structural

There is no structural consequence as a result of this event.

Thermal

There is no effect on the thermal performance of the system as a result of this event.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the lightning accident does not affect the safe operation of the HI-STAR 100 System.

11.2.11.3 Lightning Dose Calculations

An evaluation of lightning strikes demonstrates that the effect of a lightning strike has no effect on the confinement boundary or shielding materials. Therefore, no further analysis is necessary.

11.2.11.4 Lightning Accident Corrective Action

The HI-STAR 100 System will not sustain any damage from the lightning accident. There is no surveillance or corrective action required.

11.2.12 Burial Under Debris

11.2.12.1 Cause of Burial Under Debris

Burial of the HI-STAR 100 System under debris is not a credible accident. During normal storage operations at the ISFSI, there are no structures over the casks. The minimum regulatory distance of 100 meters from the ISFSI to the nearest site boundary and the controlled area around the ISFSI concrete pad precludes the close proximity of substantial amounts of vegetation.

There is no credible mechanism for the HI-STAR 100 System to become completely buried under debris. However, for conservatism, complete burial under debris is considered.

11.2.12.2 Burial Under Debris Analysis

Burial of the HI-STAR 100 System does not impose a condition that would have more severe consequences for criticality, confinement, shielding, and structural analyses than that performed for the other accidents analyzed. The debris would provide additional shielding to reduce radiation doses. The accident external pressure bounds any credible pressure loading caused by the burial under debris.

Burial under debris can affect thermal performance because the debris acts as an insulator and heat sink. This will cause the HI-STAR 100 System and fuel cladding temperatures to increase. A thermal analysis has been performed to determine the time for the fuel cladding temperatures to reach the short term accident condition temperature limits during a burial under debris accident.

To demonstrate the inherent safety of the HI-STAR 100 System, a bounding analysis which considers the debris to act as a perfect insulator is considered. Under this scenario, the contents of the HI-STAR 100 System will undergo a transient heat up under adiabatic conditions. The minimum time required for the fuel cladding to reach the short term design fuel cladding temperature limit depends on the amount of thermal inertia of the cask, the cask initial conditions, and the spent nuclear fuel decay heat generation. All three of these parameters are conservatively bounded by the values in Table 11.2.4.

Using the values stated in Table 11.2.4, the bounding cask temperature rise of less than 5°F (2.8°C) per hour is determined. This provides in excess of 60 hours of time before the cladding temperatures exceed the short term fuel cladding temperature limit.

A vertically-oriented cask containing an MPC-68 has the highest computed steady-state fuel cladding temperature. If 300°F is postulated as the permissible temperature rise the resultant pressure in the MPC cavity can be calculated as a result of the burial under debris accident.

Chapter 4 calculates the MPC internal pressure with an ambient temperature of 80°F (27°C), 100% fuel rods ruptured, full insolation, and maximum decay heat, and reports the maximum value of 84.6

psig (583 kPa) in Table 4.4.15 at an average MPC cavity temperature of 499.2°K. Using this pressure, an assumed increase in the average temperature of 300°F (499.2°K to 665.9°K), and the ideal gas law, the resultant MPC internal pressure is calculated below.

$$\frac{P_1}{P_2} = \frac{T_1}{T_2}$$

$$P_2 = \frac{P_1 T_2}{T_1}$$

$$P_2 = \frac{(84.6\,psig + 14.7)(665.9°K)}{499.2°K}$$

$$P_2 = 132.5\ psia\ or\ 117.8\ psig$$

The accident MPC internal design pressure of 200 psig or 1379 kPa (Table 2.2.1) bounds the resultant pressure calculated above. Therefore, no additional analysis is required.

Structural

The structural evaluation of the MPC enclosure vessel for normal internal pressure conditions bounds the pressure calculated above. Therefore, the resulting stresses from the normal condition internal pressure bound the stresses as a result of this event and are well within the allowable values, as discussed in Section 3.4.

Thermal

The MPC internal pressure for the burial under debris accident is calculated above. As can be seen, the 200 psig (1379 kPa) design basis internal pressure for accident conditions used in the structural evaluation bounds the calculated value for this accident.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the burial under debris accident does not affect the safe operation of the HI-STAR 100 System, if the debris is removed within 60 hours of overpack burial.

11.2.12.3 Burial Under Debris Dose Calculations

As discussed in the burial under debris analysis, the shielding is enhanced while the HI-STAR 100 System is covered. As the overpack reaches elevated temperatures, the neutron shielding material will exceed its design basis temperature. This will cause some degradation of the neutron shield effectiveness. However, the loss of neutron shield effectiveness is bounded by the assumption of complete loss of the neutron shield in the shielding analysis of the post-accident HI-STAR 100 System in Chapter 5.

The elevated temperatures will not cause the breach of the confinement system and the short term fuel cladding temperature is not exceeded. Therefore, the only radiological impact is the decreased effectiveness of the overpack neutron shield, which is bounded by the analysis in Chapter 5.

11.2.12.4 Burial Under Debris Accident Corrective Action

Analysis of the burial under debris accident shows that the fuel cladding peak temperatures will not exceed the short term limit if the debris is removed within 60 hours. Upon detection of the burial under debris accident, the ISFSI operator shall assign personnel to remove the debris with mechanical and manual means as necessary. After uncovering the overpack, the cask shall be visually and radiologically inspected for any damage.

11.2.13 Extreme Environmental Temperature

11.2.13.1 Cause of Extreme Environmental Temperature

The extreme environmental temperature is postulated as a constant ambient temperature caused by extreme weather conditions. To determine the effects of the extreme temperature, it is conservatively assumed that the temperature persists for a sufficient duration to allow the HI-STAR 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STAR 100 System, with its corresponding large thermal inertia and the limited duration for the extreme temperature, this assumption is conservative.

11.2.13.2 Extreme Environmental Temperature Analysis

The accident condition considering an environmental temperature of 125°F (52°C) for a duration sufficient to reach thermal equilibrium is evaluated with respect to accident condition design temperatures listed in Table 2.2.3. The evaluation is performed with design basis fuel with the maximum decay heat and the most restrictive thermal resistance. The 125°F (52°C) extreme environmental temperature is applied with full solar insolation.

For a vertically-oriented system, the HI-STAR 100 System maximum temperatures for components close to the design basis temperatures are listed in Tables 4.4.10 and 4.4.11. These temperatures are conservatively calculated at the normal environmental temperature of 80°F (27°C). The extreme environmental temperature is 125°F (52°C), which is an increase of 45°F (25°C). The extreme environmental condition temperatures are calculated by adding 45°F (25°C) to the maximum normal temperatures of the highest component temperature from the MPC-68 or MPC-24. Table 11.2.6 lists the component temperatures at the extreme environmental temperatures. As illustrated by the table, all the temperatures except the neutron shield are well below the accident condition design basis temperatures. The extreme environmental temperature is of a short duration (several consecutive days would be highly unlikely) and the resultant temperatures are evaluated against short-term accident condition temperature limits. Therefore, the HI-STAR 100 System will continue to operate safely under the extreme environmental temperatures.

For horizontally-oriented systems, Chapter 3 the HI-STAR 100 transport SAR considers 100°F (38°C) as the normal condition ambient temperature. As described in Section 4.4.2, however, this transport condition is bounding for normal storage conditions. Therefore, it is necessary to increase the horizontal transport temperatures by 45°F (25°C) for account for the extreme environmental temperature accident condition. Table 11.2.6 lists the component temperatures at the extreme environmental temperatures. As illustrated by the table, all the temperatures except the neutron shield in a horizontally-oriented cask are well below the accident condition design basis temperatures. The extreme environmental temperature is of a short duration (several consecutive days would be highly unlikely) and the resultant temperatures are evaluated against short-term accident condition temperature limits. Therefore, the HI-STAR 100 System will continue to operate safely under the extreme environmental temperatures.

Additionally, the extreme environmental temperature generates internal pressures which are bounded by the pressure calculated for the fire accident condition because the fire accident condition temperatures are much higher than the temperatures as a result of the extreme environmental temperature. As shown in Table 11.2.3 for the fire condition event pressures, the accident condition pressures are below the limit specified in Table 2.2.1.

<u>Structural</u>

The structural evaluation of the MPC enclosure vessel for accident condition internal pressure bounds the pressure resulting from this event. Therefore, the resulting stresses from this event are bounded by that of the accident condition and are well within the allowable values, as discussed in Section 3.4.

<u>Thermal</u>

The resulting temperatures for the system and fuel assembly cladding are provided in Table 11.2.6. As can be seen from this table, all temperatures except the neutron shield in a vertically-oriented cask are within the short-term accident condition allowable values specified in Table 2.2.3. The neutron shield temperature does exceed the long-term normal condition temperature specified in Table 2.2.3 by 16°F (8.9°C).

Shielding

The peak neutron shield temperature is higher than the stipulated the long-term normal condition temperature specified in Table 2.2.3 by 16°F (8.9°C). This extreme ambient temperature will persist for a short duration (3-day average) and therefore the degradation in the neutron shield will be negligible.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is negligible degradation in shielding and no degradation in confinement capabilities as discussed above, there is a negligible effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the extreme environmental temperature accident does not affect the safe operation of the HI-STAR 100 System.

11.2.13.3 Extreme Environmental Temperature Dose Calculations

The extreme environmental temperature may cause very localized regions of the neutron shielding material to exceed its normal design temperature for short time durations. The bulk of the neutron shield material away from these local hot spots will remain within the stipulated normal condition temperature limits. Consequently, degradation of the neutron shield effectiveness is negligible. However, the loss of neutron shield effectiveness is bounded by the assumption of complete loss of the neutron shield in the shielding analysis of the post-accident HI-STAR 100 System in Chapter 5.

The elevated temperatures will not cause a breach of the confinement system and the short-term fuel cladding temperature is not exceeded. Therefore, the only radiological impact is the decreased effectiveness of the overpack neutron shield, which is bounded by the analysis in Chapter 5.

11.2.13.4 Extreme Environmental Temperature Corrective Action

Analysis of the extreme environmental temperature accident demonstrates that the only possible consequence is a slight loss in neutron shield effectiveness. Upon detection of an extreme environmental temperature accident, the cask shall be radiologically inspected for any damage.

Table 11.2.1

SUMMARY OF TEMPERATURE-DEPENDENT FORCED CONVECTION
HEAT TRANSFER CORRELATION PARAMETERS FOR AIR

Parameter	**Trend with Increasing Temperatures**	**Criteria to Maximize h_{fc}**	**Conservative Parameter Value**	**Evaluated At**
Temperature Range	100°F-1475°F (38-800°C)	NA	NA	NA
Density	Decreases	Reynolds Number	High	100°F (38°C)
Viscosity	Increases	Reynolds Number	Low	100°F (38°C)
Conductivity (K_f)	Increases	h_{fc} Proportional to K_f	High	1475°F (800°C)

Table 11.2.2

MAXIMUM HI-STAR 100 SYSTEM TEMPERATURE UNDER
A FIRE ACCIDENT CONDITION

Component	Initial Condition (°F [°C])	During Fire (°F [°C])	Post-Fire Cooldown (°F [°C])	Short-Term Temperature Limit (°F [°C])
Fuel Cladding	741 [394]	741 [394]	771 [411]	1058 [570]
Basket Periphery	393 [201]	393 [201]	422 [217]	950 [510]
MPC Shell	331 [166]	331 [166]	364 [184]	775 [413]
Overpack Inner Shell	292 [144]	292 [144]	328 [164]	500 [260]
Overpack Closure Plate†	155 [68]	484 [251]	484 [251]	700 [371]
Overpack Top Flange	164 [73]	524 [273]	524 [273]	700 [371]
Overpack Baseplate Periphery†	197 [92]	496 [258]	496 [258]	700 [371]
Neutron Shield Inner Surface	273 [134]	273 [134]	314 [157]	††
Neutron Shield Outer Surface	233 [112]	514 [268]	551 [288]	††
Overpack Enclosure Shell	228 [109]	854 [457]	854 [457]	1000 [538]

† Overpack closure plate, overpack port plug, and overpack port cover seals short-term temperature limits are 1200°F (649°C). The maximum fire condition seals temperature is bounded by the reported closure plate and baseplate maximum temperatures. Consequently, a large margin of safety exists to permit safe operation of seals in the overpack helium retention boundary.

†† Neutron shield integrity during fire is discussed in the text.

Table 11.2.3

MAXIMUM HI-STAR 100 SYSTEM FIRE ACCIDENT CONDITION
MPC CAVITY PRESSURES†

Condition	Pressure (psig / kPa)		
	MPC-24††	**MPC-68**	**MPC-32**
During Fire Event	57.9 / 8.4	75.1 / 10.9	133.5 / 19.4
Accident Design Pressure	200 / 29	200 / 29	200 / 29

† Pressure analysis is based on NUREG-1536 criteria and a conservatively bounding 216°F (120°C) MPC cavity temperature rise.

†† PWR fuel in MPC-24 includes hypothetical BPRA rods rupture in the pressure calculations.

Table 11.2.4

SUMMARY OF INPUTS FOR ADIABATIC CASK HEAT-UP

Minimum Weight of HI-STAR 100 System (lb [kg])	200,000 [90,909]
Lower Heat Capacity of Carbon Steel (BTU/lb/°F [J/kg-K])	0.1 [417]
Initial Uniform Temperature of Cask (°F [°C])	749† [398]
Bounding Maximum Decay Heat (kW)	20

† The cask is initially conservatively assumed to be at a uniform temperature equal to temperature limit of the fuel cladding for long-term storage (see Table 4.3.1).

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Table 11.2.5

SUMMARY OF HI-STAR 100 SYSTEM MAXIMUM
POST-FIRE COOLDOWN (33 HOURS AFTER FIRE) TEMPERATURES

Location	**Temperature (°F [°C])**
Hottest MPC Basket Cross Section:	
Basket center	755 [402]
Basket periphery	419 [215]
MPC shell	358 [181]
Overpack inner shell	317 [158]
Overpack enclosure shell	249 [121]
MPC Basket Bottom:	
Basket center	285 [141]
Basket periphery	238 [114]
MPC shell	231 [111]
Overpack inner shell	225 [107]
Overpack enclosure shell	188 [87]
MPC Basket Top:	
Basket center	229 [109]
Basket periphery	199 [93]
MPC shell	193 [89]
Overpack inner shell	187 [86]
Overpack outer shell	166 [74]

Table 11.2.6

MAXIMUM TEMPERATURES CAUSED BY EXTREME ENVIRONMENTAL TEMPERATURES (°F [°C])

Temperature Location	**Normal**	**Calculated Extreme Environment**	**Accident Condition Design Temperature**
Vertically-Oriented Systems			
Fuel cladding	741† [394]	786 [419]	1058 [570]
MPC basket	725† [385]	770 [410]	950 [510]
MPC outer shell surface	332†† [167]	377 [192]	775 [410]
MPC/overpack helium gap outer surface	292†† [144]	337 [169]	500 [260]
Neutron shield inner surface	274†† [134]	319 [159]	300 [149]
Overpack shell outside surface	229†† [109]	274 [134]	1000 [535]
Horizontally-Oriented Systems			
Fuel cladding	713††† [378]	758 [403]	1058 [570]
MPC basket	697††† [369]	742 [394]	950 [510]
MPC outer shell surface	315†††† [157]	360 [182]	775 [410]
MPC/overpack helium gap outer surface	291†††† [144]	336 [169]	500 [260]
Neutron shield inner surface	271†††† [133]	316 [158]	300 [149]
Overpack shell outside surface	222†††† [106]	267 [131]	1000 [535]

† MPC-68 normal storage maximum temperatures from Table 4.4.11.

†† MPC-24 normal storage maximum temperatures from Table 4.4.10.

††† MPC-68 horizontal storage from Table 3.4.11 of [11.0.1].

†††† PWR MPCs horizontal storage from Table 3.4.10 of [11.0.1].

11.3 Regulatory Compliance

Chapter 11 has been written to provide an identification and analysis of hazards, as well as a summary of the HI-STAR 100 System's response to both off-normal and accident or design-basis events. When evaluating each event, the cause of the event, detection of the event, summary of event consequences and regulatory compliance, and corrective course of action are provided. The information provided in Chapter 11 can be summarized as follows:

- Structures, systems, and components of the HI-STAR 100 System are adequate to prevent accidents and to mitigate the consequences of accidents and natural phenomena events that do occur.
- The spacing of the HI-STAR 100 overpacks, discussed in Section 1.4 of the FSAR, will ensure accessibility of the equipment and services required for emergency response to the events evaluated in Chapter 11.
- The Technical Specifications for the HI-STAR 100 System are provided as Appendix A to Certificate of Compliance 72-1008.
- The HI-STAR 100 System has been evaluated to demonstrate that it will maintain confinement of radioactive material under credible accident conditions.
- An accident or natural phenomena event will not preclude the ready retrieval of spent fuel for further processing or disposal.
- The spent fuel will be maintained in a subcritical condition under accident conditions.
- Neither off-normal nor accident conditions will result in a dose, to an individual outside the controlled area, that exceeds the limits of 10 CFR 72.104(a) or 72.106(b), respectively.
- No instruments or control systems are required to remain operational under accident conditions.

The accident design criteria for the HI-STAR 100 System is in compliance with 10 CFR Part 72 and the accident design and acceptance criteria have been satisfied. The accident evaluation of the HI-STAR 100 System demonstrates that it will provide for safe storage of spent fuel during credible accident situations. This is based on the analyses summarized in Chapter 11, 10 CFR Part 72, appropriate regulatory guides, applicable codes and standards, and accepted engineering practice.

11.4 REFERENCES

[11.0.1] HI-STAR 100 SAR, Holtec Report HI-951251, Revision 15, Chapter 3, Docket No. 71-9261.

[11.0.2] HI-STORM FW FSAR, Holtec Report HI-2084239, Revision 1, Chapters 4 and 12, Docket No. 72-1032.

[11.0.3] HI-STORM 100 FSAR, Holtec Report HI-2002444, Revision 12, Chapters 4 and 11, Docket No. 72-1014.

[11.2.1] Chun, et al., "Dynamic Impact Effects on Spent Fuel Assemblies," Lawrence Livermore National Laboratory, UCID-21246, October 1987.

[11.2.2] ESEERCO Project EP91-29 and EPRI Project 3100-02, "Debris Collection System for Boiling Water Reactor Consolidation Equipment," B&W Fuel Company, October 1995.

[11.2.3] Jacob, M., "Heat Transfer," John Wiley & Sons, Inc. page 555, (1967).

[11.2.4] Cianos, N., and Pierce, E.T., "A Ground Lightning Environment for Engineering Usage," Technical Report No. 1, SRI Project No. 1834, Standard Research Institute, Menlo Park, CA, August 1997.

[11.2.5] Avallone, E.A., and Baumeister, T., Mark's Standard Handbook for Mechanical Engineering, Ninth Edition, McGraw Hill Inc., 1987.

[11.2.6] IAEA Safety Standards, "Regulations for the Safe Transport of Radioactive Material," International Atomic Energy Agency, Vienna, 1985.

[11.2.7] "Thermal Measurements in a Series of Large Pool Fires", Sandia Report SAND85-0196.TTC-0659.UC71, August 1987.

CHAPTER 12: OPERATING CONTROLS AND LIMITS†

Much of the new information in this chapter is directly extracted from previously NRC approved Holtec dockets; this information is shown in *italics*. In Chapter 12, this information was extracted from References [1.2.5 and 1.2.7]. All changes in this revision are marked with revision bars.

12.1 PROPOSED OPERATING CONTROLS AND LIMITS

The HI-STAR 100 System provides passive dry storage of spent fuel assemblies in interchangeable MPCs with redundant multi-pass welded closure. The loaded MPC is enclosed in a dual-purpose metal overpack. This chapter defines the operating controls and limits (i.e., Technical Specifications) including their supporting bases for deployment and storage of a HI-STAR 100 System at an ISFSI. The information provided in this chapter is in full compliance with NUREG-1536 [12.1.1].

12.1.1 NUREG-1536 (Standard Review Plan) Acceptance Criteria

12.1.1.1 This portion of the FSAR establishes the commitments regarding the HI-STAR 100 System and its use. Other 10CFR72 [12.1.2] and 10CFR20 [12.1.3] requirements in addition to the Technical Specifications may apply. The conditions for a general license holder found in 10CFR72.212 [12.1.2] shall be met by the licensee prior to spent fuel loading into the HI-STAR 100 System. The general license conditions governed by 10CFR72 [12.1.2] are not repeated with these Technical Specifications. Licensees are required to comply with all commitments and requirements.

12.1.1.2 The Technical Specifications provided herein are primarily established to maintain subcriticality, confinement boundary integrity, shielding and radiological protection, heat removal capability, and structural integrity under normal, off-normal and accident conditions. Table 12.1.1 addresses each of these conditions respectively and identifies the appropriate Technical Specification(s) designed to control the condition. Table 12.1.2 provides the list of Technical Specifications for the HI-STAR 100 System.

† Wherever multiple units are shown, the US units are the governing value, and the SI units are shown for information only.

Table 12.1.1

HI-STAR 100 SYSTEM CONTROLS

Condition to be Controlled	**Applicable Technical Specifications**
Criticality Control	Refer to Chapter 2 for fuel specifications and design features.
Confinement Boundary Integrity	2.1.1 Multi-Purpose Canister (MPC)
Shielding and Radiological Protection	Refer to Chapter 2 for fuel specifications and design features. 2.1.1 Multi-Purpose Canister (MPC) 2.1.4 Fuel Cool-Down 2.2.1 OVERPACK Average Surface Dose Rates 2.2.2 SFSC Surface Contamination
Heat Removal Capability	Refer to Chapter 2 for fuel specifications and design features. 2.1.1 Multi-Purpose Canister (MPC) 2.1.2 OVERPACK
Structural Integrity	2.1.2 OVERPACK 2.1.3 SFSC Lifting Requirements

Table 12.1.2

HI-STAR 100 TECHNICAL SPECIFICATIONS[†]

NUMBER	TECHNICAL SPECIFICATION
1.0	USE AND APPLICATION 1.1 Definitions 1.2 Logical Connectors 1.3 Completion Times 1.4 Frequency
2.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
2.1.1	Multi-Purpose Canister (MPC)
2.1.2	OVERPACK
2.1.3	SFSC Lifting Requirements
2.1.4	Fuel Cool-Down
2.2.1	OVERPACK Average Surface Dose Rates
2.2.2	SFSC Surface Contamination
3.0	ADMINISTRATIVE CONTROLS AND PROGRAMS 3.1 Training Program 3.2 Pre-Operational Testing and Training Exercise 3.3 Special Requirements for First System in Place 3.4 Radioactive Effluent Control Program

[†] Refer to Certificate of Compliance 72-1008, Appendix A for Technical Specifications and Chapter 2 for fuel specifications and design features.

12.2 DEVELOPMENT OF OPERATING CONTROLS AND LIMITS

This section provides a discussion of the operating controls and limits for the HI-STAR 100 System to assure long-term performance consistent with the conditions analyzed in this FSAR. In addition to the controls and limits provided in the Technical Specifications contained in Appendix A to Certificate of Compliance (CoC) 72-1008, the licensee shall ensure that the following training and dry run activities are performed.

12.2.1 Training Modules

Training modules are to be developed under the licensee's training program to require a comprehensive, site-specific training, assessment, and qualification (including periodic re-qualification) program for the operation and maintenance of the HI-STAR 100 Spent Fuel Storage Cask (SFSC) System and the Independent Spent Fuel Storage Installation (ISFSI). The training modules shall include the following elements, at a minimum:

1. HI-STAR 100 System Design (overview);
2. ISFSI Facility Design (overview);
3. Systems, Structures, and Components Important to Safety (overview)
4. HI-STAR 100 System Final Safety Analysis Report (overview);
5. NRC Safety Evaluation Report (overview);
6. Certificate of Compliance conditions;
7. HI-STAR 100 Technical Specifications and other Conditions for Use;
8. HI-STAR 100 Regulatory Requirements (e.g., 10CFR72.48, 10CFR72, Subpart K, 10CFR20, 10CFR73);
9. Required instrumentation and use;
10. Inspection personnel qualifications
11. Operating Experience Reviews
12. HI-STAR 100 System and ISFSI Procedures, including
    - Procedural overview
    - Fuel qualification and loading
    - MPC /overpack rigging and handling, including safe load pathways

- MPC welding operations
- Overpack closure
- Auxiliary equipment operation and maintenance (e.g., draining, vacuum drying, helium backfilling, and cooldown)
- MPC/overpack pre-operational and in-service inspections and tests
- Transfer and securing of the loaded overpack onto the transport vehicle
- Transfer and offloading of the overpack at the ISFSI
- Preparation of MPC/overpack for fuel unloading
- Unloading fuel from the MPC/overpack
- Surveillance
- Radiation protection
- Maintenance
- Security
- Off-normal and accident conditions, responses, and corrective actions

12.2.2 Dry Run Training

A dry run training exercise of the loading, closure, handling, and transfer of the HI-STAR 100 System shall be conducted by the licensee prior to the first use the system to load spent fuel assemblies. The dry run shall include, but is not limited to the following:

1. Receipt inspection of HI-STAR 100 System components.

2. Moving the HI-STAR 100 MPC/overpack into the spent fuel pool.

3. Preparation of the HI-STAR 100 System for fuel loading.

4. Selection and verification of specific fuel assemblies to ensure type conformance.

5. Locating specific assemblies and placing assemblies into the MPC (using a dummy fuel assembly), including appropriate independent verification.

6. Remote installation of the MPC lid and removal of HI-STAR 100 overpack/MPC from the spent fuel pool.

7. MPC welding, NDE inspections, pressure testing, draining, vacuum drying, helium backfilling and leakage testing *(The forced helium dehydrator may be used in lieu of the VDS).*

8. HI-STAR 100 overpack closure, draining, vacuum drying, helium backfilling and leakage testing.

9. HI-STAR 100 overpack upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling arrangement.

10. Placement of the HI-STAR 100 System at the ISFSI.

12.2.3 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings

The controls and limits apply to operating parameters and conditions which are observable, detectable, and/or measurable. The HI-STAR 100 System is completely passive during storage and requires no monitoring instruments.

12.2.4 Limiting Conditions for Operation

Limiting conditions for operation specify the minimum capability or level of performance that is required to assure that the HI-STAR 100 System can fulfill its safety functions.

12.2.4.1 Equipment

The HI-STAR 100 System and its components have been analyzed for specified normal, off-normal, and accident conditions, including extreme environmental conditions. Analysis has shown in this FSAR that no credible condition or event prevents the HI-STAR 100 System from meeting its safety function. As a result, there is no threat to public health and safety from any postulated accident condition or analyzed event. When all equipment is loaded, tested, and placed into storage in accordance with procedures developed for the ISFSI, no failure of the system to perform its safety function is expected to occur.

12.2.5 Surveillance Requirements

The analyses provided in this FSAR show that the HI-STAR 100 System fulfills its safety functions, provided that the Technical Specifications in Appendix 12.A are met. Surveillance requirements during loading, unloading, and on-site transfer operations are provided in the Technical Specifications.

12.2.6 Design Features

This section describes HI-STAR 100 System design features that are Important to Safety. These features require design controls and fabrication controls. The design features, detailed herein, are established in specifications and drawings which are controlled through the quality assurance program presented in Chapter 13. Fabrication controls and inspections to assure that the HI-STAR 100 System is fabricated in accordance with the design drawings and the requirements of this FSAR are described in Chapter 9.

12.2.6.1 MPC

a. Basket material composition, properties, dimensions, and tolerances for criticality control.

b. Canister material mechanical properties for structural integrity of the confinement boundary.

c. Canister and basket material thermal properties and dimensions for heat transfer control.

d. Canister and basket material composition and dimensions for dose rate control.

12.2.6.2 HI-STAR 100 Overpack

a. HI-STAR 100 overpack material mechanical properties and dimensions for structural integrity to provide protection of the MPC and shielding of the spent nuclear fuel assemblies during loading, unloading and handling operations.

b. HI-STAR 100 overpack material thermal properties and dimensions for heat transfer control.

c. HI-STAR 100 overpack material composition and dimensions for dose rate control.

12.3 TECHNICAL SPECIFICATIONS

Technical Specifications for the HI-STAR 100 System are provided in Appendix A to CoC 72-1008. Fuel specifications and design features are provided in Appendix B to CoC 72-1008. Bases for the Technical Specifications in CoC Appendix A are provided in FSAR Appendix 12.A. The format and content of the HI-STAR 100 System Technical Specifications and Bases are that of the Improved Standard Technical Specifications for power reactors, to the extent they apply to a dry spent fuel storage cask system. NUMARC Document 93-03, "Writer's Guide for the Restructured Technical Specifications" [12.3.9] was used as a guide in the development of the Technical Specifications and Bases.

12.4 REGULATORY EVALUATION:

Table 12.1.2 lists the Technical Specifications for HI-STAR 100 System. The Technical Specifications are detailed in Appendix A to CoC 72-1008. Fuel specifications and design features are contained in Chapter 2.

The conditions for use of HI-STAR 100 System identify necessary Technical Specifications to satisfy 10 CFR Part 72, and the applicable acceptance criteria have been satisfied. The proposed Technical Specifications, fuel specifications, and design features provide reasonable assurance that the HI-STAR 100 will allow safe storage of spent fuel and is in compliance with 10 CFR Part 72, the regulatory guides applicable codes and standards, and accepted practices.

12.5 REFERENCES

[12.1.1] U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems", NUREG-1536, Final Report, January 1997.

[12.1.2] *U.S. Code of Federal Regulations,* Title 10, "Energy", Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

[12.1.3] *U.S. Code of Federal Regulations,* Title 10, "Energy", Part 20, "Standards for Protection Against Radiation."

[12.3.1] R.W., Knoll, *et al*., Pacific Northwest Laboratory, "Evaluation of Cover Gas Impurities and Their Effects on the Dry Cask Storage of LWR Spent Fuel,"PNL-6365, DE88 003988, November 1987.

[12.3.2] American Society of Mechanical Engineers "Boiler and Pressure Vessel Code"

[12.3.3] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment," ANSI N14.5-1997.

[12.3.4] *U.S. Code of Federal Regulations,* Title 10, "Energy", Part 71, "Packaging and Transport of Radioactive Materials."

[12.3.5] NUREG-0554, Single Failure Proof Cranes for Nuclear power Plants.

[12.3.6] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 KG) or More for Nuclear Materials", ANSI N14.6, 1993.

[12.3.7] Witte, M., et al., "Evaluation of Low-Velocity Impacts Tests of Solid Steel Billet onto Concrete Pads, and Application to Generic ISFSI Storage Cask for Tipover and Side Drop." Lawrence Livermore National Laboratory, UCRL-ID-126295, Livermore, California, March 1997.

[12.3.8] American Society of Nondestructive Testing – American Society for Metals, "Nondestructive Testing Handbook, Volume One, Leakage Testing", SAN 204-7586, pp 448, June 1982.

[12.3.9] Nuclear Management and Resources Council, Inc. – "Writer's Guide for the Restructured Technical Specifications" NUMARC 93-03, February 1993.

APPENDIX 12.A

TECHNICAL SPECIFICATION BASES

FOR THE HOLTEC HI-STAR 100 SPENT FUEL STORAGE CASK SYSTEM

(47 Pages Including this Page)

BASES TABLE OF CONTENTS

B 2.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 2.0.1, 2.0.2, 2.0.4, and 2.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 2.0.1

LCO 2.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Specification).

LCO 2.0.2

LCO 2.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore a system or component or to restore variables to within specified limits. Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS. The second type of Required Action specifies the

(continued)

BASES

LCO 2.0.2 (continued)

remedial measures that permit continued operation that is not further
restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience.

LCO 2.0.3

This specification is not applicable to a dry storage cask system because it describes conditions under which a power reactor must be shut down when an LCO is not met and an associated ACTION is not met or provided. The placeholder is retained for consistency with the power reactor technical specifications.

LCO 2.0.4

LCO 2.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the HI-STORM 100 System in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Facility conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in being required to

(continued)

LCO 2.0.4 (continued)

exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continuing with dry fuel storage activities for an unlimited period of time in a specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the dry storage system. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

The provisions of LCO 2.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 2.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of an SFSC.

Exceptions to LCO 2.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 2.0.5

LCO 2.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or determined to not meet the LCO to comply with the ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 2.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of testing to demonstrate:

a. The equipment being returned to service meets the LCO; or

b. Other equipment meets the applicable LCOs.

(continued)

BASES

LCO 2.0.5 (continued)	The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed testing. This Specification does not provide time to perform any other preventive or corrective maintenance.
LCO 2.0.6	This specification is not applicable to a dry storage cask system because it describes conditions under which a power reactor must be shut down when an LCO is not met and an associated ACTION is not met or provided. The placeholder is retained for consistency with the power reactor technical specifications.
LCO 2.0.7	This specification is not applicable to a dry storage cask system because it describes conditions under which a power reactor must be shut down when an LCO is not met and an associated ACTION is not met or provided. The placeholder is retained for consistency with the power reactor technical specifications.

B 2.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 2.0.1 through SR 2.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 2.0.1

SR 2.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify that systems and components meet the LCO and variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 2.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when:

a. The systems or components are known to not meet the LCO, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the HI-STORM 100 System is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 2.0.2, prior to returning equipment to service. Upon completion of maintenance, appropriate post-maintenance testing is required. This includes ensuring applicable Surveillances

(continued)

SR 2.0.1 (continued)

are not failed and their most recent performance is in accordance with SR 2.0.2. Post maintenance testing may not be possible in the current specified conditions in the Applicability due to the necessary dry storage cask system parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow dry fuel storage activities to proceed to a specified condition where other necessary post maintenance tests can be completed.

SR 2.0.2

SR 2.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 2.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 2.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications as a Note in the Frequency stating, "SR 2.0.2 is not applicable."

As stated in SR 2.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension

(continued)

BASES

SR 2.0.2 (continued)

to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the affected equipment in an alternative manner.

The provisions of SR 2.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

SR 2.0.3

SR 2.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 2.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of HI-STORM 100 System conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified facility conditions, is discovered not to have been performed when specified, SR 2.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 2.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of changes in the specified conditions in the Applicability imposed by the Required Actions.

(continued)

SR 2.0.3 (continued)

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 2.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered to not meet the LCO or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment does not meet the LCO, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 2.0.1.

SR 2.0.4

SR 2.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.

This Specification ensures that system and component requirements and variable limits are met before entry into specified conditions in the Applicability for which these systems and components ensure safe conduct of dry fuel storage activities.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 2.0.4 restricting a change in specified condition. When a system, subsystem, division, component, device, or variable is

(continued)

SR 2.0.4 (continued)

outside its specified limits, the associated SR(s) are not required to be performed per SR 2.0.1, which states that Surveillances do not have to be performed on equipment that has been determined to not meet the LCO. When equipment does not meet the LCO, SR 2.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 2.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 2.0.4 will govern any restrictions that may (or may not) apply to specified condition changes.

The provisions of SR 2.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 2.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of an SFSC.

The precise requirements for performance of SRs are specified such that exceptions to SR 2.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

B 2.1 SFSC Integrity

B 2.1.1 Multi-Purpose Canister (MPC)

BASES

BACKGROUND

An OVERPACK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Certificate of Compliance. A lid is then placed on the MPC. The OVERPACK and MPC are raised to the top of the spent fuel pool surface. The OVERPACK and MPC are then moved into the cask preparation area where dose rates are measured and the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and vacuum drying or forced helium dehydration is performed. The MPC cavity is backfilled with helium and leakage tested. Additional dose rates are measured and the MPC vent and drain cover plates and closure ring are installed and welded. Inspections are performed on the welds. The OVERPACK lid is installed and secured. The annulus space between the MPC and OVERPACK is drained, dried and backfilled with helium gas. The OVERPACK seals are tested for leakage. Contamination measurements are completed prior to moving the OVERPACK and MPC to the ISFSI.

MPC cavity vacuum drying is utilized to remove residual moisture from the MPC fuel cavity after the MPC has been drained of water. Any water that has not drained from the fuel cavity evaporates from the fuel cavity due to the vacuum. This is aided by the temperature increase due to the temperature of the fuel and by the heat added to the MPC from the optional warming pad, if used.

After the completion of vacuum drying, the MPC cavity is backfilled with helium to a pressure greater than atmospheric pressure.

(continued)

BASES (continued)

BACKGROUND (continued)

Backfilling of the MPC fuel cavity with helium promotes gaseous heat dissipation and the inert atmosphere protects the fuel cladding. Providing a helium pressure greater than atmospheric pressure at room temperature (70^{o}F), eliminates air in-leakage over the life of the MPC because the cavity pressure rises due to heat up of the confined gas by the fuel decay heat during storage. In-leakage of air could be harmful to the fuel. Prior to moving the SFSC to the storage pad, the MPC helium leak rate is determined to ensure that the fuel is confined.

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the multiple confinement boundaries and systems. The barriers relied on are the fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the MPC in which the fuel assemblies are stored. Long-term integrity of the fuel and cladding depend on storage in an inert atmosphere. This is accomplished by removing water from the MPC and backfilling the cavity with an inert gas at a positive pressure (> 1 atm). The thermal analyses of the MPC assume that the MPC cavity is filled with dry helium.

LCO

A dry, helium filled and sealed MPC establishes an inert heat removal environment necessary to ensure the integrity of the multiple confinement boundaries. Moreover, it also ensures that there will be no air in-leakage into the MPC cavity that could damage the fuel cladding over the storage period.

APPLICABILITY

The dry, sealed and inert atmosphere is required to be in place during TRANSPORT OPERATIONS and STORAGE OPERATIONS to ensure both the confinement barriers and heat removal mechanisms are in place during these operating

(continued)

BASES

APPLICABILITY (continued)

periods. These conditions are not required during LOADING OPERATIONS or UNLOADING OPERATIONS as these conditions are being established or removed, respectively during these periods in support of other activities being performed with the stored fuel.

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent SFSCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the cavity vacuum drying pressure or demoisturizer exit gas temperature limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the potential quantity of moisture left within the MPC cavity. Since moisture remaining in the cavity during these modes of operation may represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

A.2

Once the quantity of moisture potentially left in the MPC cavity is determined, a corrective action plan shall be developed and implemented to the extent necessary to return the MPC to an analyzed condition. Since the quantity of moisture estimated

(continued)

BASES

ACTIONS

A.2 (continued)

under Required Action A.1 can range over a broad scale, different recovery strategies may be necessary. Since moisture remaining in the cavity during these modes of operation may represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to develop and complete the corrective actions commensurate with the safety significance of the CONDITION.

B.1

If the helium backfill pressure limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the helium pressure within the MPC cavity. Since too much helium in the MPC cavity during these modes represents a potential overpressure concern, an engineering evaluation shall be performed in a timely manner. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

B.2

Once the helium pressure in the MPC cavity is determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the helium pressure estimated under Required Action B.1 can range over a broad scale, different recovery strategies may be necessary. Since elevated helium pressures existing in the MPC cavity represent potential overpressure concerns, corrective actions should be developed and implemented in a timely manner. The Completion Time is sufficient to develop and complete the corrective actions commensurate with the safety significance of the CONDITION.

(continued)

BASES

ACTIONS
(continued)

C.1

DELETED

(continued)

BASES

ACTIONS (continued)

D.1

If the MPC fuel cavity cannot be successfully returned to a safe, analyzed condition, the fuel must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable based on the time required to move the OVERPACK to the cask preparation area, perform fuel cooldown operations, re-flood the MPC, cut the MPC lid welds, move the TRANSFER CASK into the spent fuel pool, remove the MPC lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 2.1.1.1, SR 2.1.1.2, and SR 2.1.1.3

The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment. Cavity dryness is demonstrated by evacuating the cavity to a very low absolute pressure and verifying that the pressure is held over a specified period of time, or verifying that the gas exiting the FHD is held at a certain temperature for a specified period of time. A low vacuum pressure or exit gas temperature is an indication that the cavity is dry. Having the proper helium backfill pressure ensures adequate heat transfer from the fuel to the fuel basket and surrounding structure of the MPC.

All three of these surveillances must be successfully performed during LOADING OPERATIONS to ensure that the conditions are established for TRANSPORT OPERATIONS and STORAGE OPERATIONS which preserve the analysis basis supporting the cask design.

REFERENCES

1. FSAR Sections 4.4, 7.2, 7.3 and 8.1

B 2.1 SFSC Integrity

B 2.1.2 OVERPACK

BASES

BACKGROUND

An OVERPACK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Certificate of Compliance. A lid is then placed on the MPC. The OVERPACK and MPC are raised to the top of the spent fuel pool surface. The OVERPACK and MPC are then moved into the cask preparation area where dose rates are measured and the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and vacuum drying is performed. The MPC cavity is backfilled with helium and leakage tested. Additional dose rates are measured and the MPC vent and drain cover plates and closure ring are installed and welded. Inspections are performed on the welds. The OVERPACK lid is installed and secured. The annulus space between the MPC and OVERPACK is drained, vacuum dried and backfilled with helium gas. The OVERPACK seals are tested for leakage. Contamination measurements are completed prior to moving the OVERPACK and MPC to the ISFSI.

Vacuum drying of the annulus between the MPC and the OVERPACK is performed to remove residual moisture from the annulus after it has been drained of water. Water that has not drained from the annulus evaporates from the annulus due to the vacuum. This is aided by the temperature increase due to the temperature of the fuel and by the heat added to the MPC from the optional warming pad, if used.

(continued)

BASES

BACKGROUND (continued)

Backfilling of the OVERPACK annulus with helium promotes heat transfer from the MPC to the OVERPACK structure. Providing a helium pressure greater than atmospheric pressure ensures that there will be no in-leakage of air over the life of the SFSC. In-leakage of air could degrade the heat transfer features of the SFSC. Prior to moving the SFSC to the storage pad, the OVERPACK annulus helium leak rate is determined to ensure that sufficient helium remains to provide adequate heat transfer.

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the multiple confinement boundaries and systems. The barriers relied on are the fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the MPC in which the fuel assemblies are stored. No confinement credit is taken for the OVERPACK boundary. Long-term integrity of the spent fuel depends on the ability of the SFSC to reject heat to the environment. This is accomplished, in part, by retaining helium in the annulus between the MPC and the OVERPACK. By removing water from the annulus, the boiling of residual water and associated pressurization of the annulus during storage at the ISFSI is avoided. Backfilling the annulus with an inert gas optimizes the ability of the SFSC to transfer heat from the MPC to the OVERPACK. In addition, the thermal analyses assume that the annulus is filled with dry helium.

LCO

A dry, helium filled and sealed OVERPACK annulus establishes an inert cooling space necessary to ensure heat rejection to the environment. Moreover, it also ensures that there will be no air in-leakage into the annulus that could negatively affect heat transfer.

(continued)

BASES (continued)

APPLICABILITY

The dry, sealed and inert atmosphere is required to be in place during TRANSPORT OPERATIONS and STORAGE OPERATIONS to ensure a heat transfer mechanism is in place during these operating periods. These conditions are not required during LOADING OPERATIONS or UNLOADING OPERATIONS as these conditions are being established or removed, respectively during these periods in support of other activities being performed with the stored MPC.

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent SFSC's that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the OVERPACK annulus vacuum drying pressure limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the potential quantity of moisture left within the annulus. Since moisture remaining in the annulus during these modes of operation may represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

A.2

Once the quantity of moisture potentially left in the OVERPACK annulus is determined, a corrective action plan shall be developed and actions completed to return the SFSC to an analyzed condition. Since the quantity of moisture estimated under Required Action A.1 can range over a broad

(continued)

ACTIONS
(continued)

A.2 (continued)

scale, different recovery strategies may be necessary. Since moisture remaining in the annulus during these modes of operation represents a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to develop and complete the corrective actions commensurate with the safety significance of the CONDITION.

B.1

If the helium backfill pressure limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the quantity of helium within the OVERPACK annulus. Since abnormal quantities of helium in the annulus during these modes represents a minimal impact, immediate action is not necessary. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

B.2

Once the quantity of helium in the annulus is determined, a corrective action plan shall be developed and initiated to the extent necessary to return the SFSC to an analyzed condition. Since the quantity of helium estimated under Required Action B.1 can range over a broad scale, different recovery strategies may be necessary. Since abnormal quantities of helium in the annulus during these modes represents a minimal impact, immediate action is not necessary. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

(continued)

BASES

ACTIONS (continued)

C.1

If the OVERPACK helium leak rate limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the potential leak rate and quantity of helium remaining within the annulus. The significance of the situation is mitigated by the existence of the MPC confinement boundary. Since abnormal leak rates from the annulus during these modes represents a minimal impact, immediate action is not necessary. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

C.2

Once the cause and consequences of the elevated leak rate from the OVERPACK are determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the recovery mechanisms can range over a broad scale, based on theevaluation performed under Required Action C.1, different recovery strategies may be necessary. Since abnormal leak rates from the annulus during these modes represents a minimal impact, immediate action is not necessary. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 2.1.2.1, SR 2.1.2.2, and SR 2.1.2.3

The long-term integrity of the stored fuel is dependent, in part, on adequate heat transfer from the stored fuel to the environment. OVERPACK annulus dryness is demonstrated by evacuating the annulus to a very low absolute pressure and verifying that the pressure is held over a specified period of time. A low vacuum pressure is an indication that the annulus is dry. Having the proper helium backfill pressure ensures adequate heat transfer from the MPC to the OVERPACK structure. Meeting the helium leak rate limit ensures there is adequate helium in the annulus for long term storage.

All three of these surveillances must be successfully performed during LOADING OPERATIONS to ensure that the conditions are established for TRANSPORT OPERATIONS and STORAGE OPERATIONS which preserve the analysis basis supporting the cask design.

REFERENCES

1. FSAR Sections 4.4, 7.2, 7.3 and 8.1

B 2.1 SFSC INTEGRITY

B 2.1.3 SFSC Lifting Requirements

BASES

BACKGROUND

A loaded SFSC is transported between the loading facility and the ISFSI using a transporter. The SFSC may be handled in either the horizontal or vertical orientation depending on the site cask handling limitations. The height to which the SFSC is lifted is limited to ensure that the structural integrity of the SFSC is not compromised should the SFSC be dropped.

For lifting of the loaded OVERPACK using devices which are integral to a structure governed by 10CFR Part 50 regulations, 10CFR50 requirements apply.

APPLICABLE SAFETY ANALYSIS

The structural analyses of the SFSC demonstrate that the drop of a loaded SFSC from the Technical Specification height limits to a surface having the characteristics described in the Appendix B to Certificate of Compliance 72-1008 will not compromise SFSC integrity or cause physical damage to the contained fuel assemblies.

LCO

Limiting the SFSC lifting height during TRANSPORT OPERATIONS maintains the operating conditions of the SFSC within the design and analysis basis. The maximum lifting height is a function of the SFSC design and the orientation that the SFSC is carried. The lifting height requirements are specified in LCO 2.1.3.a for the vertical and horizontal orientations.

Appendix B to Certificate of Compliance 72-1008 provides the characteristics of the drop surface assumed in the analyses. As required by 10 CFR 72.212(b)(3), each licensee must "...determine whether or not the reactor site parameters...are enveloped by the cask design bases..." Therefore, licensees must evaluate the storage pad and, if applicable, the site transport route to assure that they are bounded by the features specified in the CoC.

(continued)

BASES

LCO (continued)

Alternatively, LCO 2.1.3.b allows the use of lifting devices designed in accordance with NUREG-0612 and having redundant drop protection design features. If a suitably designed lifting device is used, dropping the SFSC is not considered credible, and the lift heights of LCO 2.1.3.a do not apply.

Alternatively, LCO 2.1.3.c allows for site-specific transport conditions which are not encompassed by those of LCO 2.1.3.a or 2.1.3.b. Under this alternative, the licensee shall evaluate the site-specific conditions to ensure that drop accident loads do not exceed 60 g's. This alternative analysis shall be commensurate with the analysis which forms the basis for LCO 2.1.3.a.

APPLICABILITY

The APPLICABILITY is modified by a note which states that the LCO is not applicable while the transporter is in the FUEL BUILDING or is being handling by a device providing support from underneath. The first part of the note is acceptable based on the relatively short duration of time TRANSPORT OPERATIONS take place in the FUEL BUILDING. This LCO does not apply if the SFSC is supported from underneath (e.g., air pads, heavy haul trailer or rail car) because the OVERPACK is not being lifted and a drop accident is not credible.

This LCO is applicable outside of the FUEL BUILDING during TRANSPORT OPERATIONS when the SFSC is being lifted or otherwise suspended above the surface below. This includes movement of the SFSC while suspended from a transporter (i.e., a vertical crawler). It is not applicable during STORAGE OPERATIONS since the SFSC is not considered lifted.

(continued)

BASES (continued)

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each SFSC not meeting the LCO. Subsequent SFCSs that don't meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If none of the SFSC lifting requirements are met, immediate action must be initiated and completed expeditiously to comply with one of the three lifting requirements in order to preserve the SFSC design and analysis basis.

SURVEILLANCE REQUIREMENTS

SR 2.1.3.1

The SFSC lifting requirements of LCO 2.1.3 must be verified to be met after the SFSC is suspended from, or secured in the transporter and prior to the transporter beginning to move the SFSC to or from the ISFSI. This ensures potential drop accidents during TRANSPORT OPERATIONS are bounded by the drop analyses.

For compliance with LCO 2.1.3.a, lifting heights are to be measured from the lowest surface on the OVERPACK to the potential impact surface.

REFERENCES

1. FSAR, Sections 3.4.10, 8.1, and 8.3

B 2.1 SFSC INTEGRITY

B 2.1.4 MPC Cavity Reflooding

BASES

BACKGROUND

In the event that an MPC must be unloaded, the OVERPACK with its enclosed MPC is returned to the cask preparation area to begin the process of fuel unloading. The MPC closure ring, and vent and drain port cover plates are removed. The MPC gas is sampled to determine the integrity of the spent fuel cladding. The pressure in the MPC cavity is ensured to be less than the design pressure. This is accomplished via direct measurement of the MPC gas pressure or via analysis

Following fuel cool-down, the MPC is then re-flooded with water
After ensuring the MPC cavity pressure meets the LCO limit, the MPC is then re-flooded with water at a controlled rate and/or the pressure monitored to ensure that the pressure remains below 100 psig. Once the cavity is filled with water, the MPC lid weld is removed leaving the MPC lid in place. The transfer cask and MPC are placed in the spent fuel pool and the MPC lid is removed. The fuel assemblies are removed from the MPC and the MPC and transfer cask are removed from the spent fuel pool and decontaminated.

Ensuring that the MPC cavity pressure is less than the LCO limit ensures that any steam produced within the cavity is safely vented to an appropriate location and eliminates the risk of high MPC pressure due to sudden generation of steam during re-flooding.

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the multiple confinement boundaries and systems. The barriers relied on are the fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the MPC in which the fuel assemblies are stored. Standard practice in the dry cask industry has historically been to directly reflood the cask with water. This standard practice is known not to induce fuel cladding failures.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)	The Integrity of the MPC depends on maintaining the internal cavity pressures within design limits. This is accomplished by introducing water to the cavity in a controlled manner such that there is no sudden formation of steam during MPC re-flooding. (Ref. 1).
LCO	Determining the MPC cavity pressure prior to and during re-flooding ensures that there will be sufficient venting of any steam produced to avoid excessive MPC pressurization.
APPLICABILITY	The MPC cavity pressure is controlled during UNLOADING OPERATIONS after the OVERPACK and integral MPC are back in the FUEL BUILDING and are no longer suspended from, or secured in, the transporter. Therefore, the Cask Reflood LCO does not apply during TRANSPORT OPERATIONS and STORAGE OPERATIONS. A note has been added to the APPLICABILITY for LCO 2.1.4 which states that the LCO is only applicable during wet UNLOADING OPERATIONS. This is acceptable since the intent of the LCO is to avoid uncontrolled MPC pressurization due to water flashing during re-flooding operations. This is not a concern for dry UNLOADING OPERATIONS.
ACTIONS	A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for

(continued)

BASES

ACTIONS (continued)

each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

If the MPC cavity pressure limit is not met, actions must be taken to restore the parameters to within the limits before initiating or continuing to re-flood the MPC.

Immediately is an appropriate Completion Time because it requires action to be initiated promptly and completed without delay, but does not establish any particular fixed time limit for completing the action. This offers the flexibility necessary for users to plan and implement any necessary work activities commensurate with the safety significance of the condition, which is governed by the MPC heat load.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 2.1.4.1

The integrity of the MPC is dependent on controlling the internal MPC pressure. By controlling the MPC internal pressure prior to and during re-flooding the MPC there is sufficient steam venting capacity during MPC re-flooding.

The LCO must be met on each SFSC before the initiation of MPC re-flooding operations to ensure the design and analysis basis are preserved. If the re-flood rate is limited to the bounding value given in FSAR Section 4.5 or calculated specifically for the MPC heat load then the MPC pressure must only be verified once prior to the re-flood.

If verifying the MPC pressure using direct measurement only, the SR requires checks prior to the re-flood and every hour during re-flood. The direct measurement schedule is sufficient to prevent overpressurization of the MPC cavity as the rate of pressure rise is relatively slow compared to increase in re-flood rate.

REFERENCES

1. FSAR, Sections 4.4.1, 4.5.1.1.4, and 8.3.2.

B 2.2 SFSC Radiation Protection

B 2.2.1 OVERPACK Average Surface Dose Rates

BASES

BACKGROUND	The regulations governing the operation of an ISFSI set limits on the control of occupational radiation exposure and radiation doses to the general public (Ref. 1). Occupational radiation exposure should be kept as low as reasonably achievable (ALARA) and within the limits of 10CFR Part 20. Radiation doses to the public are limited for both normal and accident conditions.
APPLICABLE SAFETY ANALYSIS	The OVERPACK average surface dose rates are not an assumption in any accident analysis, but are used to ensure compliance with regulatory limits on occupational dose and dose to the public.
LCO	The limits on OVERPACK average surface dose rates are based on the shielding analysis of the HI-STAR 100 System (Ref. 2). The limits were selected to minimize radiation exposure to the general public and maintain occupational dose ALARA to personnel working in the vicinity of the SFSCs.
APPLICABILITY	The average OVERPACK surface dose rates apply during TRANSPORT OPERATIONS and STORAGE OPERATIONS. Radiation doses during STORAGE OPERATIONS are monitored for the OVERPACK by the SFSC user in accordance with the plant-specific radiation protection program required by 10CFR72.212(b)(6).

(continued)

BASES (continued)

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each SFSC not meeting the LCO. Subsequent SFSCs that don't meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the OVERPACK average surface dose rates are not within limits, it could be an indication that a fuel assembly was inadvertently loaded into the MPC that did not meet the specifications in Appendix B of the Certificate of Compliance. Administrative verification of the MPC fuel loading, by means such as review of video recordings and records of the loaded fuel assembly serial numbers, can establish whether a mis-loaded fuel assembly is the cause of the out of limit condition. The Completion Time is based on the time required to perform such a verification.

A.2

If the OVERPACK average surface dose rates are not within limits, and it is determined that the MPC was loaded with the correct fuel assemblies, an analysis may be performed. This analysis will determine if the OVERPACK dose rates would result in the ISFSI offsite or occupational doses exceeding regulatory limits in 10 CFR Part 20 or 10 CFR Part 72.

B.1

If it is verified that the correct fuel was not loaded or that the ISFSI offsite radiation protection requirements of 10 CFR Part 20 or 10 CFR Part 72 will not be met with the OVERPACK average surface dose rates above the LCO limit, the fuel

(continued)

BASES

ACTIONS (continued)

assemblies must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable based on the time required to move the SFSC to the cask preparation area, perform fuel cooldown operations, re-flood the MPC, cut the MPC lid welds, move the SFSC into the spent fuel pool, remove the MPC lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 2.2.1.1

This SR is modified by two notes. The first note requires dose rate measurements to be taken after the MPC has been dried. This ensures that the dose rates measured are indicative of minimal shielding conditions with no shielding provided by the water in the MPC. The second note requires the OVERPACK average surface dose rates to be measured by performing this SR after receipt, and prior to storage if the OVERPACK was loaded at an off-site facility and transported to another facility for storage. This provides assurance that dose rates remain within the LCO limits after handling and transporting the OVERPACK between sites.

This SR ensures that the OVERPACK average surface dose rates are within the LCO limits prior to moving the SFSC to the ISFSI. Surface dose rates are measured approximately at the top and sides of the overpack following standard industry practices for determining average dose rates for large containers.

REFERENCES

1. 10 CFR Parts 20 and 72.
2. FSAR Sections 5.1 and 8.1.6.

B 2.2 SFSC Radiation Protection

B 2.2.2 SFSC Surface Contamination

BASES

BACKGROUND

An SFSC is immersed in the spent fuel pool in order to load the spent fuel assemblies. As a result, the surface of the SFSC may become contaminated with the radioactive material in the spent fuel pool water. This contamination is removed prior to moving the SFSC to the ISFSI in order to minimize the radioactive contamination to personnel or the environment. This allows dry fuel storage activities to proceed without additional radiological controls to prevent the spread of contamination and reduces personnel dose due to the spread of loose contamination or airborne contamination. This is consistent with ALARA practices.

APPLICABLE SAFETY ANALYSIS

The radiation protection measures implemented at the ISFSI are based on the assumption that the exterior surfaces of the SFSC's have been decontaminated. Failure to decontaminate the surfaces of theSFSC's could lead to higher-than-projected occupational doses and potential site contamination.

LCO

Removable surface contamination on the OVERPACK exterior surfaces and accessible surfaces of the MPC is limited to 1000 dpm/100 cm^2 from beta and gamma sources and 20 dpm/100 cm^2 from alpha sources. These limits are taken from the guidance in IE Circular 81-07 (Ref. 2) and are based on the minimum level of activity that can be routinely detected under a surface contamination control program using direct survey methods. Only loose contamination is controlled, as fixed contamination will not result from the SFSC loading process. Experience has shown that these limits are low enough to prevent the spread of contamination to clean areas and are significantly less than the levels which would cause significant personnel skin dose.

(continued)

BASES

LCO (continued)

LCO 2.2.2 requires removable contamination to be within the specified limits for the exterior surfaces of the OVERPACK and accessible portions of the MPC. The location and number of surface swipes used to determine compliance with this LCO are determined based on standard industry practice and the user's plant-specific contamination measurement program for objects of this size. Accessible portions of the MPC means the upper portion of the MPC external shell wall accessible after the inflatable annulus seal is removed and before the annulus shield ring is installed. The user shall determine a reasonable number and location of swipes for the accessible portion of the MPC. The objective is to determine a removable contamination value representative of the entire upper circumference of the MPC, while implementing sound ALARA practices.

APPLICABILITY

The requirements of this LCO must be met during TRANSPORT OPERATIONS and STORAGE OPERATIONS to minimize the potential for spreading contamination. Measurement of the OVERPACK and MPC surface contamination is unnecessary during UNLOADING OPERATIONS as surface contamination would have been measured prior to moving the subject TRANSFER CASK to the ISFSI.

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each TRANSFER CASK. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each TRANSFER CASK not meeting the LCO. Subsequent TRANSFER CASKs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

(continued)

BASES

ACTIONS (continued)

A.1

If the removable surface contamination of an SFSC that has been loaded with spent fuel is not within the LCO limits, action must be initiated to decontaminate the SFSC and bring the removable surface contamination within limits. The Completion Time of 7 days is appropriate given that surface contamination does not affect the safe storage of the spent fuel assemblies.

SURVEILLANCE REQUIREMENTS

SR 2.2.2.1

This SR is modified by a note which requires the SFSC surface contamination to be measured by performing this SR after receipt, and prior to storage if the OVERPACK was loaded at an off-site facility and transported to another facility for storage. This provides assurance that contamination levels remain within the LCO limits after handling and transporting the OVERPACK between sites.

This SR verifies that the removable surface contamination on the OVERPACK and accessible portions of the MPC is less than the limits in the LCO. The Surveillance is performed using smear surveys to detect removable surface contamination. The Frequency requires performing the verification during LOADING OPERATIONS in order to confirm that the SFSC can be moved to the ISFSI without spreading loose contamination.

REFERENCES

1. FSAR Sections 8.1.5 and 8.1.6.
2. NRC IE Circular 81-07.

B 2.3 SFSC Criticality Control

B 2.3.1 Boron Concentration

BASES

BACKGROUND

An OVERPACK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Certificate of Compliance. A lid is then placed on the MPC. The OVERPACK and MPC are raised to the top of the spent fuel pool surface. The OVERPACK and MPC are then moved into the cask preparation area where the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and drying is performed. The MPC cavity is backfilled with helium. Then the MPC vent and drain cover plates and closure ring are installed and welded.

For those MPCs containing PWR fuel assemblies of relatively high initial enrichment, credit is taken in the criticality analyses for boron in the water within the MPC. To preserve the analysis basis, users must verify that the boron concentration of the water in the MPC meets specified limits when there is fuel and water in the MPC. This may occur during LOADING OPERATIONS and UNLOADING OPERATIONS.

APPLICABLE SAFETY ANALYSIS

The spent nuclear fuel stored in the SFSC is required to remain subcritical (k_{eff}<0.95) under all conditions of storage. The HI-STAR 100 SFSC is analyzed to store a wide variety of spent nuclear fuel assembly types with differing initial enrichments. For all PWR fuel loaded in the MPC-32, and for relatively high enrichment PWR fuel loaded in the MPC-24, credit was taken in the criticality analyses for neutron poison in the form of soluble boron in the water within the MPC. Compliance with this LCO preserves the assumptions made in the criticality analyses regarding credit for soluble boron.

(continued)

BASES (continued)

LCO

Compliance with this LCO ensures that the stored fuel will remain subcritical with a k_{eff}<0.95, while water is in the MPC. LCOs 2.3.1.a provides the minimum concentration of soluble boron required in the MPC water for the MPC-24 for MPCs containing all INTACT FUEL ASSEMBLIES. The limits are applicable to the MPC-24 if one or more fuel assemblies to be loaded in the MPC had an initial enrichment of U-235 greater than the value in Table 2.1-2 of Appendix B to the CoC for loading with no soluble boron credit.

LCO 2.3.1.b provides the minimum concentration of soluble boron required in the MPC water for the MPC-32 for MPCs containing all INTACT FUEL ASSEMBLIES.

The LCO also requires that the minimum soluble boron concentration for the most limiting fuel assembly array/class to be stored in the same MPC be used. This means that the highest minimum soluble boron concentration limit for all fuel assemblies in the MPC applies in cases where fuel assembly array/classes are mixed in the same MPC. This ensures the assumptions pertaining to soluble boron used in the criticality analyses are preserved.

APPLICABILITY

The boron concentration LCO is applicable whenever an MPC-24 or -32 has at least one PWR fuel assembly in a storage location and water in the MPc. For the MPC-24 when all fuel assemblies to be loaded have initial enrichments less than the limit for no soluble boron credit as provided in CoC Appendix B, Table 2.1-2, the boron concentration requirement is implicitly understood to be zero.

During LOADING OPERATIONS, the LCO is applicable immediately upon the loading of the first fuel assembly in the MPC. It remains applicable until the MPC is drained of water.

During UNLOADING OPERATIONS, the LCO is applicaple when the MPC is reflooded with water after helium cooldown operations. Note that compliance with SR 2.0.4 assures that the water to be used to flood the MPC is of the correct boron concentration to ensure the LCO Is upon entering the Applicability.

(continued)

BASES (continued)

ACTIONS

A note has been added to the ACTIONS which states that for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions

A.1 and A.2

Continuation of LOADING OPERATIONS, UNLOADING OPERATIONS, or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the SFSC in compliance with the LCO. Uf the boron concentration of water in the MPC is less than its limit, all activities LOADING OPERATIONS, UNLOADING OPERATIONS, or positive reactivity additions must be suspended immediately.

A.3

In addition to immediately suspending LOADING OPERATIONS, UNLOADING OPERATIONS, or positive reactivity additions, action to restore the concentration to within the limit specified in the LCO must be initiated immediately.

One means of complying with this action is to initiate boration of the affected MPC. In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied; only that boration be initiated without delay. In order to raise the boron concentration as quickly as possible, the operator should begin boration with the best source available for existing plant conditions

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

(continued)

BASES (continued)

SUREVELLANCE REQUIREMENTS

SR 2.3.1.1

The boron concentration in the MPC water must be verified to be within the applicable limit within four hours prior to entering the Applicability of the LCO. For LOADING OPERATIONS this means within four hours of loading the first fuel assembly into the cask.

For UNLOADING OPERATIONS, this means verifying the source of borated water to be used t ore-flood the MPC within four hours of commencing re-flooding operations. This ensures that when the LCO is applicable (upon introduction water into the MPC), the LCO will be met.

Surveillance Requirement 2.3.1.1 is modified by a note which states that SR 2.3.1.1 is only required to be performed if the MPC is submerged in water or if water is to be added to, or recirculated through the MPC. This reflects the underlying premie of this SR which is to ensure, once the correct boron concentration is established, it need only be verified thereafter if the MPC is in a state where the concentration could be changed.

There is no need to re-verify the boron concentration of the water in the MPC after it is removed from the spent fuel pool unless water is to be added to, or recirculated through the MPC, because these are the only credible activities that could potentially change the boron concentration during this time. This note also prevents the interference of unnecessary sampling activities while lid welding and other MPC storage preparation activities are taking place in an elevated radiation area atop the MPC. Plant procedures should ensure that any water to be added to, or recirculated through the MPC is at a boron concentration greater than or equal to the minimum boron concentration specified in the LCO.

REFERENCES

1. FSAR Chapter 6

APPENDIX 12.B

THIS APPENDIX HAS BEEN DELETED

CHAPTER 13†: QUALITY ASSURANCE

13.0 QUALITY ASSURANCE PROGRAM

Much of the new information in this QA program description is directly extracted from previously NRC approved Holtec dockets; this information is shown in *italics*. In Chapter 13, this information was extracted from the HI-STORM UMAX Storage Docket, 72-1040, in the NRC SER for Amendment 0, published April 2, 2015. All changes in this revision are marked with revision bars.

13.0.1 Overview

This chapter provides a summary of the quality assurance program implemented for activities related to the design, qualification analyses, material procurement, fabrication, assembly, testing and use of structures, systems, and components of the *Company's dry storage/transport systems including the* HI-STAR 100 System. *This chapter is included in this FSAR to fulfill the requirements in 10 CFR 72.140 (c) (2) and 72.2(a)(1),(b).*

Important-to-safety activities related to construction and deployment of the HI-STAR 100 System are controlled under the NRC-approved Holtec Quality Assurance Program. The Holtec QA program manual *[13.0.1] is approved by the NRC [13.0.2]* under Docket 71-0784. The Holtec QA program satisfies the requirements of 10 CFR 72, Subpart G and 10 CFR 71, Subpart H. In accordance with 10 CFR 72.140(d), this approved 10 CFR 71 QA program will be applied to spent fuel storage cask activities under 10 CFR 72. The additional recordkeeping requirements of 10 CFR 72.174 are addressed in the Holtec QA program manual and must also be complied with.

The Holtec QA program is implemented through a hierarchy of procedures and documentation, listed below.

1. Holtec Quality Assurance Program Manual

2. Holtec Quality Assurance Procedures

3. a. Holtec Standard Procedures
   b. Holtec Project Procedures

Quality activities performed by others on behalf of Holtec are governed by the supplier's quality assurance program or Holtec's QA program extended to the supplier. The type and extent of Holtec QA control and oversight is specified in the procurement documents for the specific item or service being procured. The fundamental goal of the supplier oversight portion of Holtec's QA program is to provide assurance that activities

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61.

performed in support of the supply of safety-significant items and services are performed correctly and in compliance with the procurement documents.

13.0.2 Graded Approach to Quality Assurance

Holtec International uses a graded approach to quality assurance on all safety-related or important-to-safety projects. This graded approach is controlled by Holtec Quality Assurance (QA) program documents as described in Subsection 13.0.1.

NUREG/CR-6407 [13.0.3] provides descriptions of quality categories A, B and C. Using the guidance in NUREG/CR-6407, Holtec International assigns a quality category to each individual, important-to-safety component of the HI-STAR 100 System. The ITS categories assigned to the HI-STAR 100 cask components are identified in Table 2.2.6 of this FSAR. Quality categories for ancillary equipment are provided in Table 8.1.4 of this FSAR on a generic basis. Quality categories for other equipment needed to deploy the HI-STAR 100 System at a licensee's ISFSI are defined on a case-specific basis considering the component's design function *using the guidelines of NUREG/CR-6407 [13.0.3].*

Activities affecting quality are defined by the purchaser's procurement contract for use of the HI-STAR 100 System at an independent spent fuel storage installation (ISFSI) under the general license provisions of 10CFR72, Subpart K. These activities include any or all of the following: design, procurement, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair and monitoring of HI-STAR 100 structures, systems, and components (SSCs) that are important-to-safety.

The quality assurance program described in the QA Program Manual fully complies with the requirements of 10CFR72 Subpart G and the intent of NUREG-1536 [13.0.4]. However, NUREG-1536 does not explicitly address incorporation of a QA program manual by reference. Therefore, invoking the NRC-approved QA program in this FSAR constitutes a literal deviation from NUREG-1536. This deviation is acceptable since important-to-safety activities are implemented in accordance with the latest revision of the Holtec QA program manual and implementing procedures. Further, incorporating the QA Program Manual by reference in this FSAR avoids duplication of information between the implementing documents and the FSAR and any discrepancies that may arise from simultaneous maintenance to the two program descriptions governing the same activities, and is consistent with all other Holtec storage dockets.

13.1 REFERENCES

[13.0.1] Holtec International Quality Assurance Program, Latest Approved Revision on Docket 71-0784.

[13.0.2] NRC QA Program Approval for Radioactive Material Packages No. 0784, Docket 71-0784.

[13.0.3] NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," February 1996.

[13.0.4] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," Rev 1, USNRC, 2010.