

# **FINAL SAFETY ANALYSIS REPORT**

## **ON**

# **THE HI-STORM FW MPC STORAGE SYSTEM**

By

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**Certificate of Compliance (1032) and Final Safety Analysis Report Matrix**

<b>HI-STORM FW Final Safety Analysis Report (FSAR) Revision</b>	<b>NRC Certificate of Compliance (CoC) 1032 Amendment No.</b>
<b>0</b>	<b>0</b>
<b>1</b>	<b>See Note 1 Below</b>
<b>2</b> <small>Note 2</small>	<b>0</b>
<b>3</b>	<b>1</b>
<b>4</b>	<b>1 Rev 1</b>

**Notes:**

- 1) Revision 1 of the HI-STORM FW FSAR contains the safety analyses of MPC-37 and MPC-89 in support of the FW system's original certification as well as LAR# 1 and RAI#1. This Revision 1 is part of UMAX submittal under docket # 72-1040 for configuration control.
- 2) Revision 2 of the HI-STORM FW FSAR contains all ECO/72.48 changes. Revision 0 of the FW FSAR was the basis for creating Revision 2. If the chapter in R2 doesn't contain any ECO/72.48, then the revision is kept at R0 since LAR# 1 and RAI# 1 changes are not included in Revision 2.
- 3) Revision 3 of the HI-STORM FW FSAR contains the changes related to Amendment 1. Revision 3 uses Revision 2 as the basis with all changes from Revision 2 marked with revision bars. Using this strategy, changes already submitted in Revision 1, with docket 72-1040 may be marked as changes in this Revision 3, since they are changes from Revision 2. This ensures configuration control for docket 72-1032.
- 4) Revision 4 of the HI-STORM FW FSAR contains changes related to Amendment 1 Revision 1 and ECO/72.48 changes. Revision 4 uses Revision 3 as the basis with all changes from Revision 3 marked with revision bars.

FSAR SECTION REVISION STATUS, LIST OF AFFECTED SECTIONS AND REVISION SUMMARY		
FSAR Report No.: HI-2114830		FSAR Revision Number: 4
FSAR Title:	Final Safety Analysis Report on the HI-STORM FW System	
<p>This FSAR is submitted to the USNRC in support of Holtec International's application to secure a CoC under 10CFR Part 72.</p> <p>FSAR review and verification are controlled at the chapter level and changes are annotated at the chapter level.</p> <p>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.</p> <p>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.</p>		
<b>Chapter 1 (including Glossary and Notation)</b>		
Affected Section or Table No.	Current Revision No.	Summary Description of Change
Glossary	4	The section has been revised per ECO 5018-32
Subsection 1.0.1		Updated the list of ECOs and 72.48 Evaluations for changes that are incorporated into this revision of the FSAR.
Table 1.0.3		The table has been revised per ECO 5018-32.
Paragraph 1.2.1.2		The section has been revised per ECO 5018-31.
Subparagraph 1.2.1.4.1		The section has been revised per ECO 5018-33 and Amendment 2 Rev.1.
Paragraph 1.2.1.5		The section has been revised per ECO 5018-32.
Tables 1.2.8a and 1.2.8b		The tables have been added per Amendment 1 Rev.1.
Table 1.2.9		The table has been added per ECO 5018-33.

Section 1.5		The drawing revision numbers have been updated to reflect the current drawing included in the FSAR.
Section 1.6		The section has been revised per Amendment 1 Rev. 1.
Chapter 2		
Section or Table No.	Current Revision No.	Summary Description of Change
Tables 2.0.2, 2.0.6, and 2.2.2	4	The tables have been revised per ECO 5018-22.
Table 2.1.1		The table has been revised per ECO 5018-21.
Table 2.1.2		The table has been revised per Amendment 1 Rev.1.
Subsection 2.2.3		The section was revised per ECO 5018-32.
Table 2.2.6		The table was revised per ECO 5018-32.
Chapter 3		
Section or Table No.	Current Revision No.	Summary Description of Change
Tables 3.2.3, 3.2.4, 3.2.8, 3.4.5, and 3.4.16, Paragraph 3.4.4.1, Subparagraph 3.4.4.1.3.vi, and Appendix 3.A	4	Incorporated changes from ECO 5018-21.
Table 3.2.1		The table has been revised per ECO 5018-28.
Table 3.2.2		The table has been revised per ECO 5018-21.
Paragraph 3.1.2.1		The section has been revised per ECO 5018-32.
Table 3.3.7		The table has been revised per ECO 5018-22.
Table		The table has been deleted per ECO 5018-38.

3.4.13		
Subparagraph 3.4.4.1.4		The section has been revised per ECO 5018-38.
Figures 3.4.15A and 3.4.15B		The figure has been revised per ECO 5018-38.
Figure 3.4.36		The figure has been deleted per ECO 5018-38.
Tables 3.1.9, 3.3.4, and 3.4.2		The tables have been revised per ECO 5018-42.
Paragraph 3.4.3.2		The section has been revised per ECO 5018-31.
<b>Chapter 4</b>		
<b>Section or Table No.</b>	<b>Current Revision No.</b>	<b>Summary Description of Change</b>
Paragraphs 4.4.1.iii.h and 4.4.4.4		The sections have been revised per ECO 5018-31.
Paragraph 4.4.1.9		The section has been added per ECO 5018-31.
Table 4.4.14		The table has been added per ECO 5018-31.
Tables 4.6.2 and 4.6.5		The tables have been revised per ECO 5018-31.
Tables 4.2.2 and 4.2.4		The tables have been revised per ECO 5018-33.
Paragraph 4.4.1.8		The section has been revised per ECO 5018-33.
Paragraph 4.5.4.6 and 4.5.4.7		The sections have been added per ECO 5018-33.
Tables 4.5.13, 4.5.14, and 4.5.15		The tables have been added per ECO 5018-33.
Table 4.4.6		The table has been revised per ECO 5018-41.
Subsections and Paragraphs 4.4.1, 4.4.1.1,		The sections have been revised per ECO 5018-21.

4.4.1.3, 4.4.4.1, 4.4.5.2, and 4.6.2.5	4	
Paragraphs 4.4.1.7 and 4.5.4.5		The sections have been added per ECO 5018-21.
Tables 4.4.1, 4.4.3, 4.5.3, 4.5.4, and 4.6.6		The tables have been revised per ECO 5018-21.
Tables 4.4.12, 4.4.13, 4.5.11, 4.5.12		The tables have been added per ECO 5018-21.
Paragraph 4.5.4.8		The section has been added per ECO 5018-22.
Table 4.5.9		The table has been added per ECO 5018-22.
Chapter 5		
Section or Table No.	Current Revision No.	Summary Description of Change
Subsection 5.1.1	4	The section has been revised per ECO 5018-34.
Chapter 6		
Section or Table No.	Current Revision No.	Summary Description of Change
Tables 6.1.1, 6.2.3, and 6.4.10	4	The tables have been revised per Amendment 1 Rev.1.
Paragraph 6.4.4.1		The section has been revised per ECOs 5018-34 and 5018-41.
Chapter 7 Changes		
Section or Table No.	Current Revision No.	Summary Description of Change
Section 7.2	4	The section has been revised per ECO 5018-27.
Chapter 8 Changes		

Section or Table No.	Current Revision No.	Summary Description of Change
Subsection 8.2.2	4	The section has been revised per ECO 5018-22.
Table 8.1.2		The table has been revised per ECO 5018-31.
Section 8.5		The section has been revised per ECOs 5018-38 and 5018-43.
Paragraphs 8.2.1.2.ii and 8.4.4.1		The sections have been revised per Amendment 1 Rev 1.
Chapter 9 Changes		
Section or Table No.	Current Revision No.	Summary Description of Change
Subparagraphs 9.2.6.15.e, 9.2.6.15.e.1, and 9.2.6.15.f	4	The sections have been revised per ECO 5018-29.
Subparagraphs 9.2.6.15.e.9		The section has been added per ECO 5018-29.
Tables 9.1.1 and 9.2.1		The tables have been revised per ECO 5018-32.
Chapter 10 Changes		
Section or Table No.	Current Revision No.	Summary Description of Change
Subparagraph and Paragraph 10.1.1.4 and 10.1.6.2	4	The section has been revised per ECO 5018-38 and Amendment 1 Rev. 1.
Subsection 10.1.3		The section has been revised per Amendment 1 Rev. 1.
Tables 10.1.6, 10.1.7, and 10.1.8		The tables have been revised per Amendment 1 Rev. 1.
Paragraph 10.1.2.1		The section has been revised per ECO 5018-32.
Chapter 11 Changes		
Section or Table No.	Current Revision No.	Summary Description of Change

	3	No change.
<b>Chapter 12 Changes</b>		
<b>Section or Table No.</b>	<b>Current Revision No.</b>	<b>Summary Description of Change</b>
Table 12.2.1	4	The table has been revised per ECO 5018-32.
<b>Chapter 13 Changes</b>		
<b>Section or Table No.</b>	<b>Current Revision No.</b>	<b>Summary Description of Change</b>
	3	No change.
<b>Chapter 14 Changes</b>		
<b>Section or Table No.</b>	<b>Current Revision No.</b>	<b>Summary Description of Change</b>
	0	No change.

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## GLOSSARY

**ALARA** is an acronym for As Low As Reasonably Achievable

**Ancillary or Ancillary Equipment** is the generic name of a device used to carry out short term operations.

**Bottom Lid** means the removable lid that fastens to the bottom of the HI-TRAC VW transfer cask body to create a gasketed barrier against in-leakage of pool water in the space around the MPC.

**BWR** is an acronym for Boiling Water Reactor.

**CG** is an acronym for center of gravity.

**Commercial Spent Fuel or CSF** refers to nuclear fuel used to produce energy in a commercial nuclear power plant.

**Confinement Boundary** is the outline formed by the all-welded cylindrical enclosure of the MPC shell, MPC baseplate, MPC lid, MPC port cover plates, and the MPC closure ring which provides redundant sealing.

**Confinement System** means the 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.

**Cooling Time (or post-irradiation cooling time)** for a spent fuel assembly is the time between its final discharge from the reactor to the time it is loaded into the MPC.

**Critical Characteristic** means a feature of a component or assembly that is necessary for the proper safety function of the component or assembly. Critical characteristics of a material are those attributes that have been identified, in the associated material specification, as necessary to render the material's intended function.

**DAS** is the abbreviation for the Decontamination and Assembly Station. It means the location where the Transfer Cask is decontaminated and the MPC is processed (i.e., where all operations culminating in lid and closure ring welding are completed).

**DBE** means Design Basis Earthquake.

**DCSS** is an acronym for Dry Cask Storage System.

**Damaged Fuel Assembly** is a fuel assembly with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not replaced with dummy fuel rods, missing structural components such as grid spacers, whose structural integrity has been impaired such that geometric rearrangement of fuel or gross failure of the cladding is expected based on engineering evaluations, 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 (or Canister) or DFC** means a specially designed enclosure for damaged fuel or fuel debris which permits flow of gaseous and liquid media while minimizing dispersal of gross particulates.

**Design Basis Load (DBL)** is a loading which bounds one or more events that are applicable to the storage system during its service life.

**Design Heat Load** is the computed heat rejection capacity of the HI-STORM system with a certified MPC loaded with CSF stored in uniform storage with the ambient at the normal temperature and the peak cladding temperature (PCT) limit at 400°C. The Design Heat Load is less than the thermal capacity of the system by a suitable margin that reflects the conservatism in the system thermal analysis.

**Design Life** is the minimum duration for which the component is engineered to perform its intended function set forth in this SAR, if operated and maintained in accordance with this SAR.

**Design Report** is a document prepared, reviewed and QA validated in accordance with the provisions of 10CFR72 Subpart G. The Design Report shall demonstrate compliance with the requirements set forth in the Design Specification. A Design Report is mandatory for systems, structures, and components designated as Important to Safety. The SAR serves as the Design Report for the HI-STORM FW System.

**Design Specification** is a document prepared in accordance with the quality assurance requirements of 10CFR72 Subpart G to provide a complete set of design criteria and functional requirements for a system, structure, or component, designated as Important to Safety, intended to be used in the operation, implementation, or decommissioning of the HI-STORM FW System. The SAR serves as the Design Specification for the HI-STORM FW System.

**Enclosure Vessel (or MPC Enclosure Vessel)** means the pressure vessel defined by the cylindrical shell, baseplate, port cover plates, lid, closure ring, and associated welds that provides confinement for the contents within the MPC. The Enclosure Vessel (EV) and the fuel basket together constitute the multi-purpose canister.

**Equivalent (or Equal) Material** is a material with critical characteristics (see definition above) that meet or exceed those specified for the designated material.

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**Fracture Toughness** is a property which is a measure of the ability of a material to limit crack propagation under a suddenly applied load.

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

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

**Fuel Building** is the generic term used to denote the building in which the fuel loading and where part of “short-term operations” will occur. The Fuel Building is a Part 50 controlled structure.

**Fuel Debris** is ruptured fuel rods, severed rods, loose fuel pellets, containers or structures that are supporting these loose fuel assembly parts, or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

**Fuel Spacer or Shim** is a metallic part interposed in the space between the fuel and the MPC cavity at either the top or the bottom (or both) ends of the fuel to minimize the axial displacement of the SNF within the MPC due to longitudinal inertia forces.

**High Burnup Fuel, or HBF** is a commercial spent fuel assembly with an average burnup greater than 45,000 MWD/MTU.

**HI-TRAC VW transfer cask or HI-TRAC VW** means the transfer cask used to house the MPC during MPC fuel loading, unloading, drying, sealing, and on-site transfer operations to a HI-STORM storage overpack or HI-STAR storage/transportation overpack. The HI-TRAC shields and protects the loaded MPC.

**HI-STORM overpack** or storage overpack means the cask that receives and contains the sealed multi-purpose canisters containing spent nuclear fuel for long term storage. It provides the gamma and neutron shielding, ventilation passages, missile protection, and protection against natural phenomena and accidents for the loaded MPC.

**HI-STORM FW System** consists of any loaded MPC model placed within the HI-STORM FW overpack.

**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.

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**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.

**License Life** means the duration for which the system is authorized by virtue of its certification by the U.S. NRC.

**Long-term Storage** means the time beginning after on-site handling is complete and the loaded overpack is at rest in its designated storage location on the ISFSI pad.

**Lowest Service Temperature (LST)** is the minimum metal temperature of a part for the specified service condition.

**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<sup>®</sup>** is a trade name for an aluminum/boron carbide composite neutron absorber material qualified for use in the MPCs and in wet storage applications.

**METAMIC-HT** is the trade name for the metal matrix composite made by imbedding nanoparticles of aluminum oxide and fine boron carbide powder on the grain boundaries of aluminum resulting in improved structural strength properties at elevated temperatures.

**METCON<sup>™</sup>** is a trade name for the HI-STORM overpack structure. The trademark is derived from the **metal-concrete** composition of the HI-STORM overpack.

**MGDS** is an acronym for Mined Geological Disposal System.

**Minimum Enrichment** is the minimum assembly average enrichment. Axial blankets are not considered in determining minimum enrichment.

**Moderate Burnup Fuel, or MBF** is a commercial spent fuel assembly with an average burnup less than or equal to 45,000 MWD/MTU.

**Multi-Purpose Canister or MPC** means the sealed canister consisting of a honeycombed fuel basket for spent nuclear fuel storage, contained in a cylindrical canister shell (the MPC Enclosure Vessel). There are different MPCs with different fuel basket geometries for storing PWR or BWR fuel, but all MPCs have identical exterior diameters. The MPC is the confinement boundary for storage conditions.

**MPC Transfer** means transfer of the MPC between the overpack and the transfer cask which begins when the MPC is lifted off the HI-TRAC bottom lid and ends when the MPC is supported from beneath by the overpack (or the reverse).

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**NDT** is an acronym for Nil Ductility Transition Temperature, which is defined as the temperature at which the fracture stress in a material with a small flaw is equal to the yield stress in the same material if it had no flaws.

**Neutron Absorber** is a generic term to indicate any neutron absorber material qualified for use in the HI-STORM FW System.

**Neutron Shielding** means a material used to thermalize and capture neutrons emanating from the radioactive spent nuclear fuel.

**Non-Fuel Hardware** is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs), Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), Control Element Assemblies (CEAs), Neutron Source Assemblies (NSAs), water displacement guide tube plugs, orifice rod assemblies, Instrument Tube Tie Rods (ITTRs), vibration suppressor inserts, and components of these devices such as individual rods.

**Planar-Average Initial Enrichment** is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

**Plain Concrete** is concrete that is unreinforced.

**Post-Core Decay Time (PCDT)** is synonymous with cooling time.

**PWR** is an acronym for pressurized water reactor.

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

**Regionalized Fuel Storage** is a term used to describe an optimized fuel loading strategy wherein the storage locations are ascribed to distinct regions each with its own maximum allowable specific heat generation rate.

**Removable Shielding Girdle** is an ancillary designed to be installed to provide added shielding to the personnel working in the top region of the transfer cask.

**SAR** is an acronym for Safety Analysis Report.

**Service Life** means the duration for which the component is reasonably expected to perform its intended function, if operated and maintained in accordance with the provisions of this FSAR. Service Life may be much longer than the Design Life because of the conservatism inherent in the codes, standards, and procedures used to design, fabricate, operate, and maintain the component.

**Short-term Operations** means those normal operational evolutions necessary to support fuel loading or fuel unloading operations. These include, but are not limited to MPC cavity drying, helium backfill, MPC transfer, and onsite handling of a loaded HI-TRAC VW transfer cask or HI-STORM FW overpack.

**Single Failure Proof** in order for a lifting device or special lifting device to be considered single failure proof, the design must follow the guidance in NUREG-0612, which requires that a single failure proof device have twice the normal safety margin. This designation can be achieved by either providing redundant devices (load paths) or providing twice the design factor as required by the applicable code.

**SNF** is an acronym for spent nuclear fuel.

**SSC** is an acronym for Structures, Systems and Components.

**STP** is Standard Temperature and Pressure conditions.

**TAL** is an acronym for the Threaded Ancor Location. TALs are used in the HI-STORM FW and HI-TRAC VW casks as well as the MPCs.

**Thermo-siphon** is the term used to describe the buoyancy-driven natural convection circulation of helium within the MPC fuel basket.

**Traveler** means the set of sequential instructions used in a controlled manufacturing program to ensure that all required tests and examinations required upon the completion of each significant manufacturing activity are performed and documented for archival reference.

**Undamaged 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. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as Intact Fuel Assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the fuel rod(s).

**Uniform Fuel Loading** is a fuel loading strategy where any authorized fuel assembly may be stored in any fuel storage location, subject to other restrictions in the CoC, such as those applicable to non-fuel hardware, and damaged fuel containers.

**ZPA** is an acronym for zero period acceleration.

**ZR** means any zirconium-based fuel cladding material authorized for use in a commercial nuclear power plant reactor. Any reference to Zircaloy fuel cladding in this FSAR applies to any zirconium-based fuel cladding material.



# CHAPTER 1: GENERAL DESCRIPTION

## 1.0 GENERAL INFORMATION

This final safety analysis report (FSAR) describes the Holtec International HI-STORM FW System and contains the necessary information and analyses to support a United States Nuclear Regulatory Commission (USNRC) licensing review as a spent nuclear fuel (SNF) dry storage cask under the provisions of 10 CFR 72 [1.0.1]. This report, prepared pursuant to 10 CFR 72.230, describes the basis for NRC approval and issuance of a Certificate of Compliance (CoC) on the HI-STORM FW System under 10 CFR 72, Subpart L to safely store spent nuclear fuel (SNF) at an Independent Spent Fuel Storage Installation (ISFSI) under the general license authorized by 10 CFR 72, Subpart K.

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]. The only deviation in the format from the formatting instruction in Reg. Guide 3.61 is the insertion of a chapter (Chapter 8) on material compatibility pursuant to ISG-15 and renumbering of all subsequent chapters. Rev 1A of NUREG 1536, available only as a draft document at the time of the initial composition of this report (Rev 0), has also been consulted to insure conformance.

The purpose of this chapter is to provide a general description of the design features and storage capabilities of the HI-STORM FW System, drawings of the structures, systems, and components (SSCs), designation of their safety classification, 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 that is similar in objective and scope.

Table 1.0.1 provides the principal components of the HI-STORM FW System. An MPC (containing either PWR or BWR fuel) is placed inside the HI-STORM FW overpack for long term storage. The overpack provides shielding, allows for convective cooling, and protects the MPC. The HI-TRAC VW transfer cask is used for MPC transfer and also provides shielding and protection while the MPC is being prepared for storage.

Table 1.0.2 provides a matrix of the topics in NUREG-1536 and Regulatory Guide 3.61, the corresponding 10 CFR 72 requirements, and a reference to the applicable report section that addresses each topic.

The HI-STORM FW FSAR is in full compliance with the intent of all regulatory requirements listed in Section III of each chapter of NUREG-1536. However, an exhaustive review of the provisions in NUREG-1536, particularly Section IV (Acceptance Criteria) and Section V (Review Procedures) has identified certain minor deviations in the method of compliance. Table 1.0.3 lists these deviations, along with a discussion of the approach for compliance, and justification. The justification may be in the form of supporting analysis, established industry practice, or other NRC guidance documents. Each chapter in this FSAR provides a clear statement with respect to the extent of compliance to the NUREG-1536 provisions. (The extent of compliance with NUREG-1536 in this docket mirrors that in Docket No. 72-1014.)

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The Glossary contains a listing of the terminology and notation used in this FSAR.

The safety evaluations in this FSAR are intended to bound the conditions that exist in the vast majority of domestic power reactor sites and potential away-from-reactor storage sites in the contiguous United States. This includes the potential fuel assemblies which will be loaded into the system and the environmental conditions in which the system will be deployed. 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 bases and safety analyses documented herein. In accordance with 10CFR72, Subpart K, site-specific implementation of the generically certified HI-STORM FW System requires that the licensee perform a site-specific evaluation, as defined in 10CFR72.212. The HI-STORM FW System FSAR identifies a number of conditions that are 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 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 canister transfer 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 9 and 10, and 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.

In presenting the bounding generic analyses of this safety report, selected conditions are drawn from

authoritative sources such as Regulatory Guides and NUREGs, where available. For example, the wind and tornado characteristics are excerpted from Reg. Guide 1.76 [1.0.4].

For analyses that do not have a prescribed acceptance limit or bounding condition, illustrative calculations are carried out with a fuel type most commonly used at reactor sites. The Reference SNF for PWR and BWR fuel types are listed in Table 1.0.4. These Reference SNF assemblies are used when fixed limits for compliance are not established by regulations, such as dose rates.

Where the analysis must demonstrate compliance with a fixed limit, such as the reactivity limit of 0.95 in criticality analysis, the most limiting fuel type is used in the analysis. The Design Basis Fuel (Table 2.1.4) may differ depending on the analysis being performed (e.g., thermal, structural, etc...). Thus, broadly speaking, the analyses in this FSAR belong to two categories:

- a. Those that are performed to satisfy a specific set of hard limits in the regulations or the Standard Review Plan.
- b. Those that are representative in nature and intended to demonstrate the acceptability of the analysis models and capability of the system.

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. Similarly, the following deci-numeric convention is used in the organization of chapters:

- a. A chapter is identified by a whole numeral, say *m* (i.e., *m*=3 means Chapter 3).
- b. A section is identified by one decimal separating two numerals. Thus, Section 3.1 is a section in Chapter 3.
- c. A subsection has three numerals separated by two decimals. Thus, Subsection 3.2.1 is a subsection in Section 3.2.
- d. A paragraph is denoted by four numerals separated by three decimals. Thus, Paragraph 3.2.1.1 is a paragraph in Subsection 3.2.1.
- e. A subparagraph has five numerals separated by four decimals. Thus, Subparagraph 3.2.1.1.1 is a part of Paragraph 3.2.1.1.

Tables and figures associated with a section are placed after the text narrative. Complete sections are replaced if any material in the section is changed. The specific changes are appropriately annotated. Drawing packages are controlled separately within the Holtec QA program and have individual revision numbers. If a drawing is revised in support of the current FSAR revision, that drawing is included in Section 1.5 at its latest revision level. Upon issuance of the CoC, drawings and text matter in this FSAR may be revised between formal updates under the 10CFR 72.48 process. All changes to the FSAR including the drawings are subject to a rigorous configuration control under the Company's QA program.

### 1.0.1 Engineering Change Orders

The changes authorized by the Holtec ECOs (with corresponding 10CFR72.48 evaluations, if applicable) listed in the following table are reflected in this Revision of the FSAR.

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

<b>Affected Item</b>	<b>ECO Number</b>	<b>72.48 Evaluation or Screening Number</b>
MPC-89 Basket	101-9	1053
	101-17	1112
	101-18R1	1110R1
MPC-37 Basket	102-8R1	1053
	102-18	1110
MPC Enclosure Vessel	101-8R1	1053
	101-9R1	1053
	101-12	1076
	101-13	1053
	101-15	1093
	101-16	1103
	101-17	1112
	102-8R3	1053
	102-10R1	1053
	102-11R1	1076
	102-13	1091
	102-14	1091
	102-15	1098
	102-17R1	1103
HI-STORM FW Overpack	100-7R1	1068
	100-8	1080
	100-10	1086
	100-11	1089
	100-13	1106
	100-14	N/A
HI-TRAC VW	103-11	1064
	103-12	1098
	103-14	1114
General FSAR Changes	5018-21R1	1070
	5018-22R1	1050R1
	5018-26	1053
	5018-27	1067
	5018-28	1076
	5018-29	1080
	5018-31	1086
	5018-32R3	1095R3
	5018-33R2	1103R1

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## LIST OF ECO'S AND APPLICABLE 10CFR72.48 EVALUATIONS (Cont'd)

Affected Item	ECO Number	72.48 Evaluation or Screening Number
General FSAR Changes (Cont'd)	5018-34	N/A
	5018-38R1	1110R1
	5018-41	1112
	5018-42	1114
	5018-43	N/A

TABLE 1.0.1

## HI-STORM FW SYSTEM COMPONENTS

<b>Item</b>	<b>Designation (Model Number)</b>
Overpack	HI-STORM FW
PWR Multi-Purpose Canister	MPC-37
BWR Multi-Purpose Canister	MPC-89
Transfer Cask	HI-TRAC VW

TABLE 1.0.2				
REGULATORY COMPLIANCE CROSS REFERENCE MATRIX				
Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI- STORM FW FSAR	
<b>1. General Description</b>				
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	
<b>2. Principal Design Criteria</b>				
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	10CFR72.122(b)	2.2	
		10CFR72.122(c)	2.2.3	
		10CFR72.122(b)(1)	2.2	
		10CFR72.122(b)(2)	2.2.3	
		10CFR72.122(h)(1)	2.0	
2.2.1 Tornado and Wind Loading	2.III.2.b External Conditions	10CFR72.122(b) (2)	2.2.3	
2.2.2 Water Level (Flood)	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b)(2)	2.2.3	

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REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI- STORM FW FSAR
2.2.3 Seismic	2.III.3.b Structural	10CFR72.102(f) 10CFR72.122(b)(2)	2.2.3
2.2.4 Snow and Ice	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b)	2.2.1
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)	1.5
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)	11.0, 9.0
		10CFR72.236(h)	9.0
		10CFR72.24(1)(2)	1.2.1, 1.2.2
		10CFR72.236(1)	2.3.2.1
		10CFR72.24(e) 10CFR72.104(b)	12.0, 9.0
	2.III.3.g Acceptance Tests & Maintenance	10CFR72.122(1) 10CFR72.236(g) 10CFR72.122(f) 10CFR72.128(a)(1)	10.0
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
	2.III.3.c Thermal	10CFR72.236(f)	2.3.2.
	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.126(a) 10CFR72.128(a)(2)	2.3.5
		10CFR72.128(a) (3)	2.3.2
		10CFR72.236(d)	2.3.2, 2.3.5
		10CFR72.236(e)	2.3.2

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TABLE 1.0.2			
REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI- STORM FW 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)	11.4.1
		10CFR72.24(d) 10CFR72.106(b) 10CFR72.236(d)	11.4.2
		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
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)
<b>3. Structural Evaluation</b>			
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	10CFR72.24(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	10CFR72.24(c)(3)	3.3
	3.V.2.c Structural Materials		
NA	3.III.2 Radiation, Shielding, Confinement, and Subcriticality	10CFR72.24(d) 10CFR72.124(a) 10CFR72.236(c) 10CFR72.236(d) 10CFR72.236(1)	3.4.4 3.4.7 3.4.10

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REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI- STORM FW FSAR
NA	3.III.3 Ready Retrieval	10CFR72.122(f) 10CFR72.122(h) 10CFR72.122(l)	3.4.4
NA	3.III.4 Design-Basis Earthquake	10CFR72.24(c) 10CFR72.102(f)	3.4.7
NA	3.III.5 20 Year Minimum Design Length	10CFR72.24(c) 10CFR72.236(g)	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)	3.4.4
3.4.5 Cold	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b)	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
4.2 Summary of Thermal Properties of Materials	4.V.4.b Material Properties	--	4.2
4.3 Specifications for Components	4.IV Acceptance Criteria ISG-11, Revision 3	10CFR72.122(h)(1)	4.3
4.4 Thermal Evaluation for Normal Conditions of Storage	4.IV Acceptance Criteria ISG-11, Revision 3	10CFR72.24(d) 10CFR72.236(g)	4.4, 4.5
NA	4.IV Acceptance Criteria for off-normal and accident conditions	10CFR72.24(d) 10CFR72.122(c)	4.6
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

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REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI- STORM FW FSAR
5.2 Source Specification	5.V.2 Radiation Source Definition	--	5.2
5.2.1 Gamma Source	5.V.2.a Gamma Source	--	5.2.1
5.2.2 Neutron Source	5.V.2.b Neutron Source	--	5.2.2
5.3 Model Specification	5.V.3 Shielding Model Specification	--	5.3
5.3.1 Description of the Radial and Axial Shielding Configurations	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.	--	Appendix 5.A
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 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, 6.3.3, 6.4.4 to 6.4.9
6.4.3 Criticality Results	6.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.124 10CFR72.236(c)	6.1

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TABLE 1.0.2				
REGULATORY COMPLIANCE CROSS REFERENCE MATRIX				
Regulatory Guide 3.61 Section and Content		Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI- STORM FW FSAR
6.5	Critical Benchmark Experiments	6.V.4.c Benchmark Comparisons	--	6.5, Appendix 6.A, 6.4.3
6.6	Supplemental Data	6.V.5 Supplemental Info.	--	Appendix 6.B
<b>7. Confinement</b>				
7.1	Confinement Boundary	7.III.1 Description of Structures, Systems and Components Important to Safety ISG-18	10CFR72.24(c)(3) 10CFR72.24(1)	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
7.1.2	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 ISG-18	10CFR72.24(d) 10CFR72.236(1)	7.1
7.2.1	Release of Radioactive Material	7.III.6 Release of Nuclides to the Environment	10CFR72.24(1)(1)	7.1
		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 ISG-18	10CFR72.104(a)	7.1
7.2.2	Pressurization of Confinement Vessel	--	--	7.1
7.3	Confinement Requirements for Hypothetical Accident Conditions	7.III.7 Evaluation of Confinement System ISG-18	10CFR72.24(d) 10CFR72.122(b) 10CFR72.236(l)	7.1
7.3.1	Fission Gas Products	--	--	7.1
7.3.2	Release of Contents	ISG-18	--	7.1
NA		--	10CFR72.106(b)	7.1

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TABLE 1.0.2			
REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI- STORM FW FSAR
7.4 Supplemental Data	7.V Supplemental Info.	--	--
8. Material Evaluation			
NA	X.5.1 General Considerations (ISG-15)	10CFR72.24(c)(3) 10CFR72.236(m) 10CFR72.122(a) 10CFR72.104(a) 10CFR72.106(b) 10CFR72.124 10CFR72.128(a)(2)	8.1
	X.5.2 Materials Selection (ISG-15)	10CFR72.236(m) 10CFR72.122(a) 10CFR72.104(a) 10CFR72.106(b) 10CFR72.124 10CFR72.128(a)(2) 10CFR72.122(a) 10CFR72.122(b) 10CFR72.122(c) 10CFR72.236(g) 10CFR72.236(l) 10CFR72.236(h)	8.2, 8.3, 8.4, 8.5, 8.6, 8.7, 8.9, 8.10, 8.11
	X.5.3 Chemical and Galvanic Reactions (ISG-15)	10CFR72.236(m) 10CFR72.122(a) 10CFR72.122(b) 10CFR72.122(c) 10CFR72.236(h) 10CFR72.122(h)(1) 10CFR72.236(m)	8.12
	X.5.4 Cladding Integrity (ISG-15) (ISG-11)	10CFR72.236(m) 10CFR72.122(a) 10CFR72.122(b) 10CFR72.122(c) 10CFR72.24(c)(3) 10CFR72.236(g) 10CFR72.236(h)	8.13
9. Operating Procedures			
8.1 Procedures for Loading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	9.0 et. seq.
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	9.2

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Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI- STORM FW FSAR
	8.III.3 Radioactive Effluent Control	10CFR72.24(1)(2)	9.2
	8.III.4 Written Procedures	10CFR72.212(b)(9)	9.2
	8.III.5 Establish Written Procedures and Tests	10CFR72.234(f)	9.2
	8.III.6 Wet or Dry Loading and Unloading Compatibility	10CFR72.236(h)	9.2
	8.III.7 Cask Design to Facilitate Decon	10CFR72.236(i)	9.2, 9.4
8.2 Procedures for Unloading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	9.4
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	9.4
	8.III.3 Radioactive Effluent Control	10CFR72.24(1)(2)	9.4
	8.III.4 Written Procedures	10CFR72.212(b) (9)	9.0
	8.III.5 Establish Written Procedures and Tests	10CFR72.234(f)	9.0
	8.III.6 Wet or Dry Loading and Unloading Compatibility	10CFR72.236(h)	9.0
	8.III.8 Ready Retrieval	10CFR72.122(1)	9.4
8.3 Preparation of the Cask	--	--	9.3.2
8.4 Supplemental Data	--	--	Tables 9.1.1
NA	8.III.9 Design to Minimize Radwaste	10CFR72.24(f) 10CFR72.128(a)(5)	9.2, 9.4
	8.III.10 SSCs Permit Inspection, Maintenance, and Testing	10CFR72.122(f)	Table 9.1.6
<b>10. Acceptance Criteria and Maintenance Program</b>			
9.1 Acceptance Criteria	9.III.1.a Preoperational Testing & Initial Operations	10CFR72.24(p)	9.1, 10.1
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.24(c) 10CFR72.122(a)	10.1
	9.III.1.d Test Program	10CFR72.162	10.1

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Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI- STORM FW FSAR
	9.III.1.e Appropriate Tests	10CFR72.236(1)	10.1
	9.III.1.f Inspection for Cracks, Pinholes, Voids and Defects	10CFR72.236(j)	10.1
	9.III.1.g Provisions that Permit Commission Tests	10CFR72.232(b)	10.1 <sup>(2)</sup>
9.2 Maintenance Program	9.III.1.b Maintenance	10CFR72.236(g)	10.2
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.122(f) 10CFR72.128(a)(1)	10.2
	9.III.1.h Records of Maintenance	10CFR72.212(b)(8)	10.2
NA	9.III.2 Resolution of Issues Concerning Adequacy of Reliability	10CFR72.24(i)	<sup>(3)</sup>
	9.III.1.d Submit Pre-Op Test Results to NRC	10CFR72.82(e)	<sup>(4)</sup>
	9.III.1.i Casks Conspicuously and Durably Marked	10CFR72.236(k)	10.1.7, 10.1.1.(12)
	9.III.3 Cask Identification		
11. Radiation Protection			
10.1 Ensuring that Occupational Exposures are as Low as Reasonably Achievable (ALARA)	10.III.4 ALARA	10CFR20.1101 10CFR72.24(e) 10CFR72.104(b) 10CFR72.126(a)	11.1
10.2 Radiation Protection Design Features	10.V.1.b Design Features	10CFR72.126(a)(6)	11.2
10.3 Estimated Onsite Collective Dose Assessment	10.III.2 Occupational Exposures	10CFR20.1201 10CFR20.1207 10CFR20.1208 10CFR20.1301	11.3
N/A	10.III.3 Public Exposure	10CFR72.104 10CFR72.106	11.4
	10.III.1 Effluents and Direct Radiation	10CFR72.104	
12. Accident Analyses			

TABLE 1.0.2			
REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI- STORM FW FSAR
11.1 Off-Normal Operations	11.III.2 Meet Dose Limits for Anticipated Events	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	12.1
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	12.1
	11.III.7 Instrumentation and Control for Off- Normal Condition	10CFR72.122(i)	12.1
11.2 Accidents	11.III.1 SSCs Important to Safety Designed for Accidents	10CFR72.24(d)(2) 10CFR72.122b(2) 10CFR72.122b(3) 10CFR72.122(d) 10CFR72.122(g)	12.2
	11.III.5 Maintain Confinement for Accident	10CFR72.236(1)	12.2
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	12.2, 6.0
	11.III.3 Meet Dose Limits for Accidents	10CFR72.24(d)(2) 10CFR72.24(m) 10CFR72.106(b)	12.2, 5.1.2, 7.3
	11.III.6 Retrieval	10CFR72.122(l)	9.4
	11.III.7 Instrumentation and Control for Accident Conditions	10CFR72.122(i)	(5)
NA	11.III.8 Confinement Monitoring	10CFR72.122h(4)	7.1.4
<b>13. Operating Controls and Limits</b>			
12.1 Proposed Operating Controls and Limits	--	10CFR72.44(c)	13.0
	12.III.1.e Administrative Controls	10CFR72.44(c)(5)	13.0
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 E 10CFR72 Subpart F	13.0



TABLE 1.0.2			
REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
Regulatory Guide 3.61 Section and Content	Associated NUREG- 1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI- STORM FW FSAR
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)(1)	Appendix 13.A
12.2.2 Limiting Conditions for Operation	12.III.1.b Limiting Controls	10CFR72.44(c)(2)	Appendix 13.A
	12.III.2.a Type of Spent Fuel	10CFR72.236(a)	Appendix 13.A
	12.III.2.b Enrichment		
	12.III.2.c Burnup		
	12.III.2.d Minimum Acceptance Cooling Time		
	12.III.2.f Maximum Spent Fuel Loading Limit		
	12.III.2g Weights and Dimensions		
	12.III.2.h Condition of Spent Fuel		
	12.III.2e Maximum Heat Dissipation	10CFR72.236(a)	Appendix 13.A
	12.III.2.i Inerting Atmosphere Requirements	10CFR72.236(a)	Appendix 13.A
12.2.3 Surveillance Specifications	12.III.1.c Surveillance Requirements	10CFR72.44(c)(3)	Chapter 13
12.2.4 Design Features	12.III.1.d Design Features	10CFR72.44(c)(4)	Chapter 13
12.2.4 Suggested Format for Operating Controls and Limits	--	--	Appendix 13.A
NA	12.III.2 SSC 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

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TABLE 1.0.2			
REGULATORY COMPLIANCE CROSS REFERENCE MATRIX			
<b>Regulatory Guide 3.61 Section and Content</b>	<b>Associated NUREG- 1536 Review Criteria</b>	<b>Applicable 10CFR72 or 10CFR20 Requirement</b>	<b>HI- STORM FW FSAR</b>
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)	9.0, 11.1
NA	12.III.2 Wet or Dry Loading	10CFR72.236(h)	9.0
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, 10.0
<b>14. Quality Assurance</b>			
13.1 Quality Assurance	13.III Regulatory Requirements	10CFR72.24(n) 10CFR72.140(d)	14.0
	13.IV Acceptance Criteria	10CFR72, Subpart G	

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 10.
- (3) Not applicable to HI-STORM FW 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-STORM FW System. No monitoring is required for accident conditions.
- “—” 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**  
**ALTERNATIVES TO NUREG-1536**

<b>NUREG-1536 Guidance</b>	<b>Alternate Method to Meet NUREG-1536 Intent</b>	<b>Justification</b>
2.V.2.(b)(3)(f) "10CFR Part 72 identifies several other natural phenomena events (including seiche, tsunami, and hurricane) that should be addressed for spent fuel storage."	A site-specific safety analysis of the effects of seiche, tsunami, and hurricane on the HI-STORM FW system must be performed prior to use if these events are applicable to the site.	In accordance with NUREG-1536, 2.V.(b)(3)(f), if seiche, tsunami, and hurricane are not addressed in the FSAR and they prove to be applicable to the site, a safety analysis is required prior to approval for use of the DCSS under either a site-specific, or general license.
3.V.1.d.i.(2)(a), page 3-11, "Drops with the axis generally vertical should be analyzed for both the conditions of a flush impact and an initial impact at a corner of the cask..."	The HI-STORM system components are lifted and handled by lifting equipment that meet the applicable provisions in NUREG-0612 and ANSI N14.6, as required, to preclude an uncontrolled lowering of the load.	All lifting and handling devices are also required to meet the ANSI or applicable code provisions to render the potential of a drop event in the part 72 jurisdiction non-credible.
3.V.2.b.i.(1), Page 3-19, Para. 1, "All concrete used in storage cask system ISFSIs, and subject to NRC review, should be reinforced..."	HI-STORM FW, like HI-STORM 100, uses plain concrete. The structural function is rendered by a double wall shell of carbon steel. The primary steel shell structure is designed to meet ASME Section III, Subsection NF stress limits for all normal service conditions.	Concrete is provided in the HI-STORM overpack primarily for the purpose of radiation shielding, the reinforcement in the concrete will only serve to create locations of micro-voids that will increase the emitted dose from the cask. Appendix 1.D of the HI-STORM 100 FSAR which provides technical and placement requirements on plain concrete is also invoked for HI-STORM FW concrete.
4.V.5.c, Page 4-10, Para. 3 "free volume calculations should account for thermal expansion of the cask internal components and the fuel when subjected to accident temperatures.	All free volume calculations use nominal Confinement Boundary dimensions, but the volume occupied by the fuel assemblies is calculated using maximum weights and minimum densities.	Calculating the volume occupied by the fuel assemblies using maximum weights and minimum densities conservatively over predicts the volume occupied by the fuel and correspondingly under predicts the remaining free volume.

**TABLE 1.0.3**  
**ALTERNATIVES TO NUREG-1536**

<b>NUREG-1536 Guidance</b>	<b>Alternate Method to Meet NUREG-1536 Intent</b>	<b>Justification</b>
7.V.4 “Confinement Analysis. Review the applicant’s confinement analysis and the resulting annual dose at the controlled area boundary.”	No confinement leakage analysis is performed and no effluent dose at the controlled area boundary is calculated.	<p>The MPC uses redundant closures to assure that there is no release of radioactive materials under all credible conditions. Analyses presented in Chapters 3 and 11 demonstrate that the Confinement Boundary does not degrade under all normal, off-normal, and accident conditions. Multiple inspection methods are used to verify the integrity of the Confinement Boundary (e.g., non-destructive examinations and pressure testing).</p> <p>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.</p> <p>Pursuant to ISG-18, the Holtec MPC is constructed in a manner that precludes leakage from the Confinement Boundary. Therefore, no analysis of leakage from confinement is required.</p>
13.III, “ the application must include, at a minimum, a description that satisfies the requirements of 10 CFR Part 72, Subpart G, ‘Quality Assurance’...”	Chapter 14 incorporates the NRC-approved Holtec International Quality Assurance Program Manual by reference.	The NRC has approved the Holtec Quality Assurance Program Manual under 10 CFR 71 (NRC QA Program Approval for Radioactive Material Packages No. 0784, Rev. 3). Pursuant to 10 CFR 72.140(d), Holtec will apply this QA program to all important-to-safety dry storage cask activities.

TABLE 1.0.4	
REFERENCE SNF DESIGNATIONS	
<b>Fuel Type</b>	<b>Fuel ID</b>
PWR	W 17x17
BWR	GE 10x10

## 1.1 INTRODUCTION TO THE HI-STORM FW SYSTEM

This section and the next section (Section 1.2) provide the necessary information on the HI-STORM FW System pursuant to 10CFR72 paragraphs 72.2(a)(1),(b); 72.122(a),(h)(1); 72.140(c)(2); 72.230(a),(b); and 72.236(a),(c),(h),(m).

HI-STORM (acronym for Holtec International Storage Module) FW System is a spent nuclear fuel storage system designed to be in full compliance with the requirements of 10CFR72. The model designation "FW" denotes this as a system which has been specifically engineered to withstand sustained Flood and Wind.

The HI-STORM FW System consists of a sealed metallic multi-purpose canister (MPC) contained within an overpack constructed from a combination of steel and concrete. The design features of the HI-STORM FW components are intended to simplify and reduce the on-site SNF loading and handling work effort, to minimize the burden of in-use monitoring, to provide utmost radiation protection to the plant personnel, and to minimize the site boundary dose.

The HI-STORM FW System can safely store either PWR or BWR fuel assemblies, in the MPC-37 or MPC-89, respectively. The MPC is identified by the maximum number of fuel assemblies it can contain in the fuel basket. The MPC external diameters are identical to allow the use of a single overpack design, however the height of the MPC, as well as the overpack and transfer cask, are variable based on the SNF to be loaded.

Figure 1.1.1 shows the HI-STORM FW System with two of its major constituents, the MPC and the storage overpack, in a cut-away view. The MPC, shown partially withdrawn from the storage overpack, is an integrally welded pressure vessel designed to meet the stress limits of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB [1.1.1]. The MPC defines the Confinement Boundary for the stored spent nuclear fuel assemblies. The HI-STORM FW storage overpack provides structural protection, cooling, and radiological shielding for the MPC.

The HI-STORM FW overpack is equipped with thru-wall penetrations at the bottom of the overpack and in its lid to permit natural circulation of air to cool the MPC and the contained SNF. The HI-STORM FW System is autonomous inasmuch as it provides SNF and radioactive material confinement, radiation shielding, criticality control and passive heat removal independent of any other facility, structures, or components at the site. The surveillance and maintenance required by the plant's staff is minimized by the HI-STORM FW System since it is completely passive and is composed of proven materials. The HI-STORM FW System can be used either singly or as an array at an ISFSI. The site for an ISFSI can be located either at a nuclear reactor facility or an away-from-a-reactor (AFR) location.

The information presented in this report is intended to demonstrate the acceptability of the HI-STORM FW System for use under the general license provisions of Subpart K by meeting the criteria set forth in 10CFR72.236.

The HI-STORM FW overpack is designed to possess certain key elements of flexibility to achieve

ALARA. For example:

- The HI-STORM FW overpack is stored at the ISFSI pad in a vertical orientation, which helps minimize the size of the ISFSI and leads to an effective natural convection cooling flow around the exterior and also in the interior of the MPC.
- The HI-STORM FW overpack handling operations do not require the cask to be downended at any time which eliminates the associated handling risks and facilitates compliance with radiation protection objectives.
- The HI-STORM FW overpack can be loaded with the MPC containing SNF using the HI-TRAC VW transfer cask and prepared for storage while inside the 10CFR50 [1.1.2] facility. From the 10CFR50 facility the loaded overpack is then moved to the ISFSI and stored in a vertical configuration. The overpack can also be directly loaded using the HI-TRAC VW transfer cask adjacent to the ISFSI storage pad. Some examples of MPC transfer between the FW overpack and the HI-TRAC VW transfer cask are illustrated in Figures 1.1.2 (transfer at the cask transfer facility) and 1.1.3 (transfer in the plant's egress (truck/rail) bay).

The HI-STORM FW overpack features an inlet and outlet duct configuration engineered to mitigate the sensitivity of wind direction on the thermal performance of the system. More specifically, the HI-STORM FW overpack features a radially symmetric outlet vent (located in its lid) pursuant to Holtec's Patent Number 7,330,526B2 and inlet ducts arranged at 45-degree intervals in the circumferential direction to approximate an axisymmetric opening configuration, to the extent possible.

A number of design measures are taken in the HI-STORM FW System to limit the fuel cladding temperature rise under a most adverse flood event (i.e., one that is just high enough to block the inlet duct):

- a. The overpack's inlet duct is narrow and does not allow a direct pathway through the overpack, therefore the MPC stands directly on the overpack's baseplate. This allows floodwater to come in immediate contact with the bottom of the MPC and assist the ventilation air flow in cooling the MPC.
- b. The overpack's inlet duct is tall and the MPC stands directly on the overpack's baseplate, which is welded to the overpack's inner and outer shells. Thus, if the flood water rises high enough to block air flow through the inlet ducts, substantial surface area of the lower region of the MPC will be submerged in the water. Although heat transfer from the exterior of the MPC through air circulation is limited in such a scenario, the reduction is offset by convective cooling through the floodwater itself.
- c. The MPCs are equipped with internal thermosiphon capability, which brings the heat emitted by the fuel back to the bottom region of the MPC as the circulating helium flows along the downcomer space around the fuel basket. This thermosiphon action places the heated helium in close thermal communication with the floodwater, further enhancing convective cooling via the floodwater.



The above design features of the HI-STORM FW System are subject to intellectual property protection rights (patent rights) under United States Patent and Trademark Office (USPTO) regulations.

Regardless of the storage cell count, the construction of the MPC is fundamentally the same; the basket is a honeycomb structure comprised of cellular elements. This is positioned within a circumscribing cylindrical canister shell. The egg-crate construction and cell-to-canister shell interface employed in the MPC basket impart the structural stiffness necessary to satisfy the limiting load conditions discussed in Chapter 2. Figures 1.1.4 and 1.1.5 provide cross-sectional views of the PWR and BWR fuel baskets, respectively. Figures 1.1.6 and 1.1.7 provide isometric perspective views of the PWR and BWR fuel baskets, respectively.

The HI-TRAC VW transfer cask is required for shielding and protection of the SNF during loading and closure of the MPC and during movement of the loaded MPC from the cask loading area of a nuclear plant spent fuel pool to the storage overpack. Figure 1.1.8 shows a cut away view of the transfer cask. The MPC is placed inside the HI-TRAC VW transfer cask and moved into the cask loading area of nuclear plant spent fuel pools for fuel loading (or unloading). The HI-TRAC VW/MPC assembly is designed to prevent (contaminated) pool water from entering the narrow annular space between the HI-TRAC VW and the MPC while the assembly is submerged. The HI-TRAC VW transfer cask also allows dry loading (or unloading) of SNF into the MPC in a hot cell.

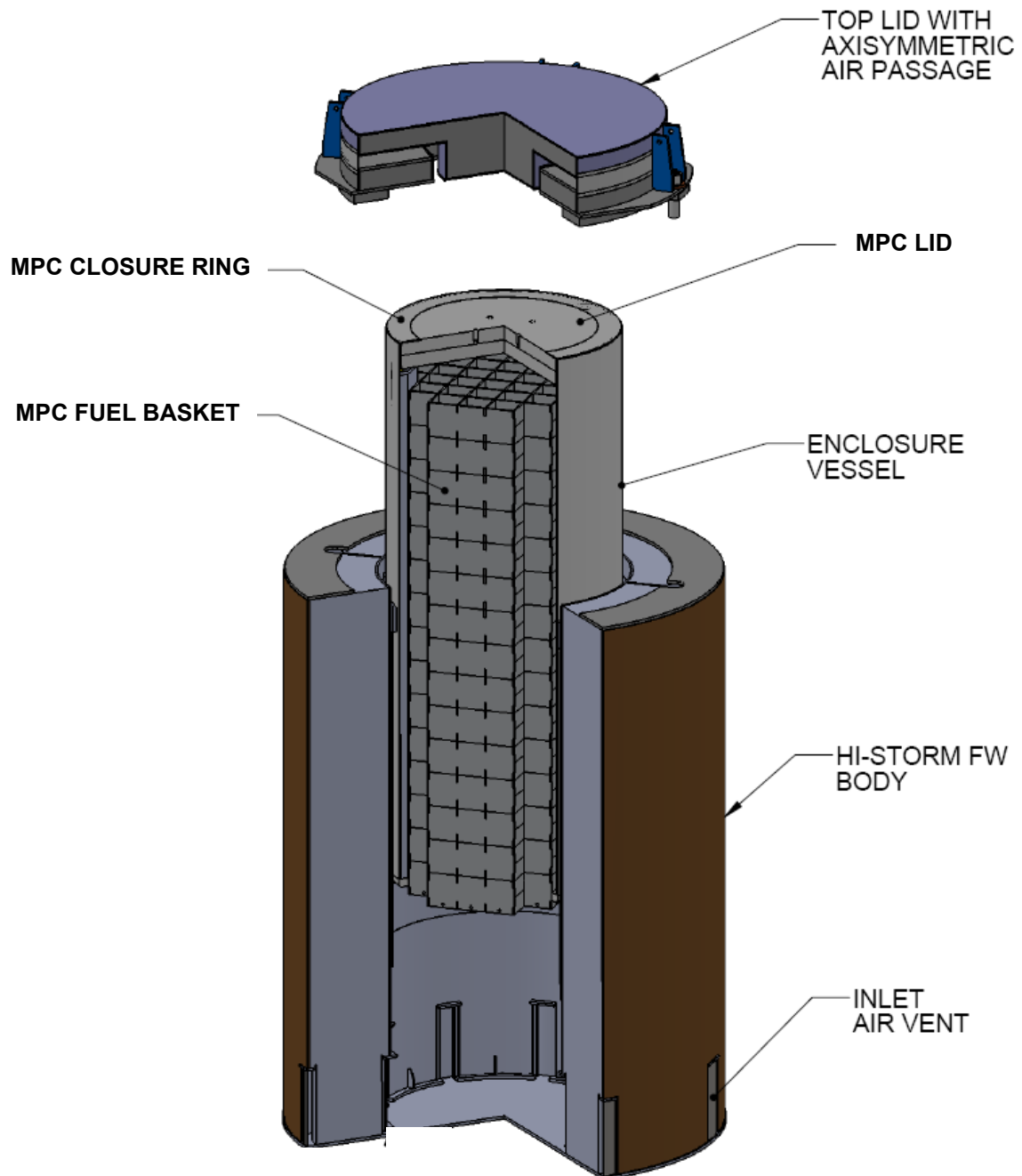
To summarize, the HI-STORM FW System has been engineered to:

- maximize shielding and physical protection for the MPC;
- maximize resistance to flood and wind;
- minimize the extent of handling of the SNF;
- minimize dose to operators during loading and handling;
- require minimal ongoing surveillance and maintenance by plant staff;
- facilitate SNF transfer of the loaded MPC to a compatible transport overpack for transportation;
- permit rapid and unencumbered decommissioning of the ISFSI;

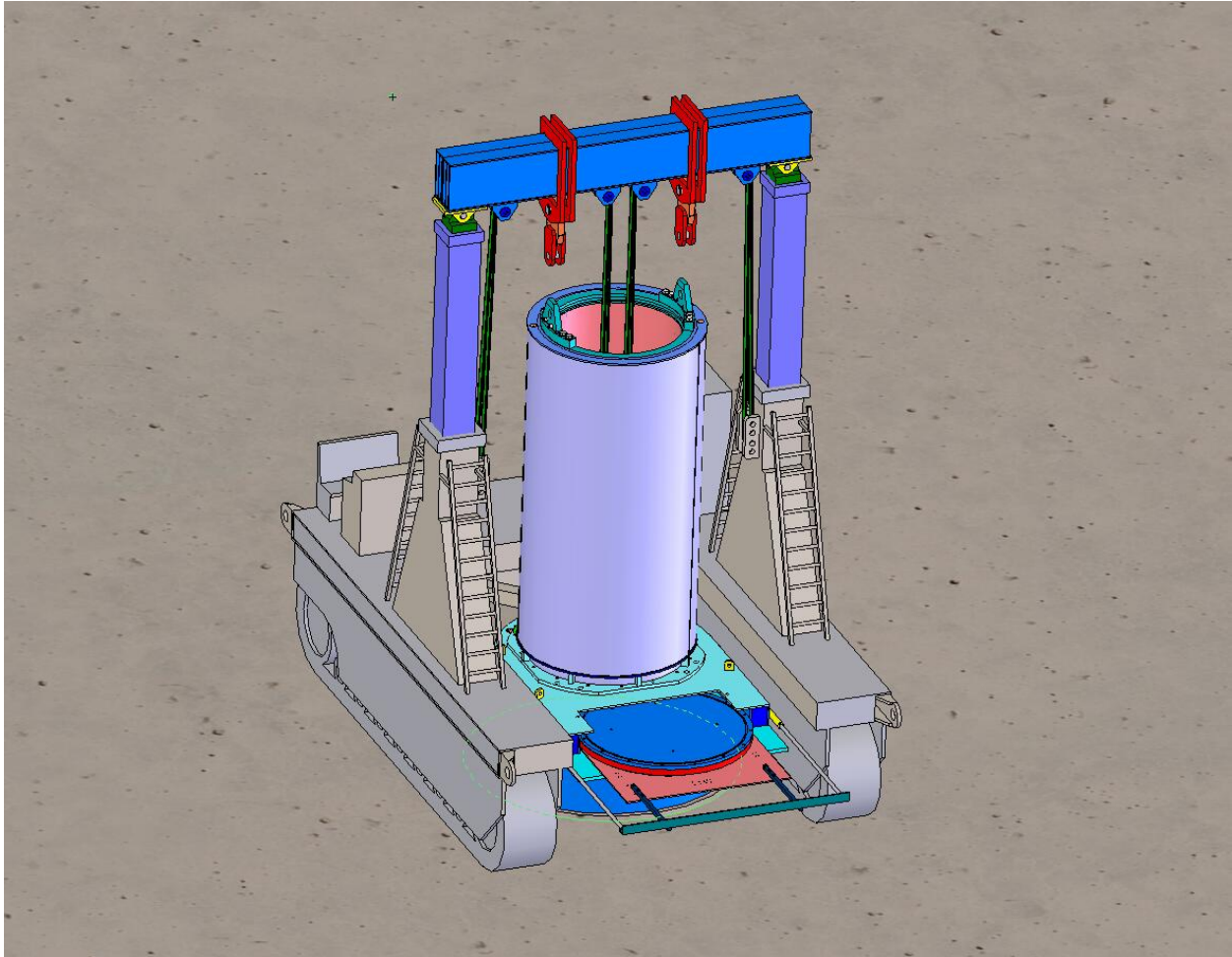
Finally, design criteria for a forced helium dehydration (FHD) system, as described in Appendix 2.B of the HI-STORM 100 FSAR [1.1.3] is compatible with HI-STORM-FW. Thus, the references to a FHD system in this FSAR imply that its design criteria must comply with the provisions in the latest revision of the HI-STORM 100 FSAR (Docket No. 72-1014).

All HI-STORM FW System components (overpack, transfer cask, and MPC) are designated ITS and their sub-components are categorized in accordance with NUREG/CR-6407 [1.1.4].

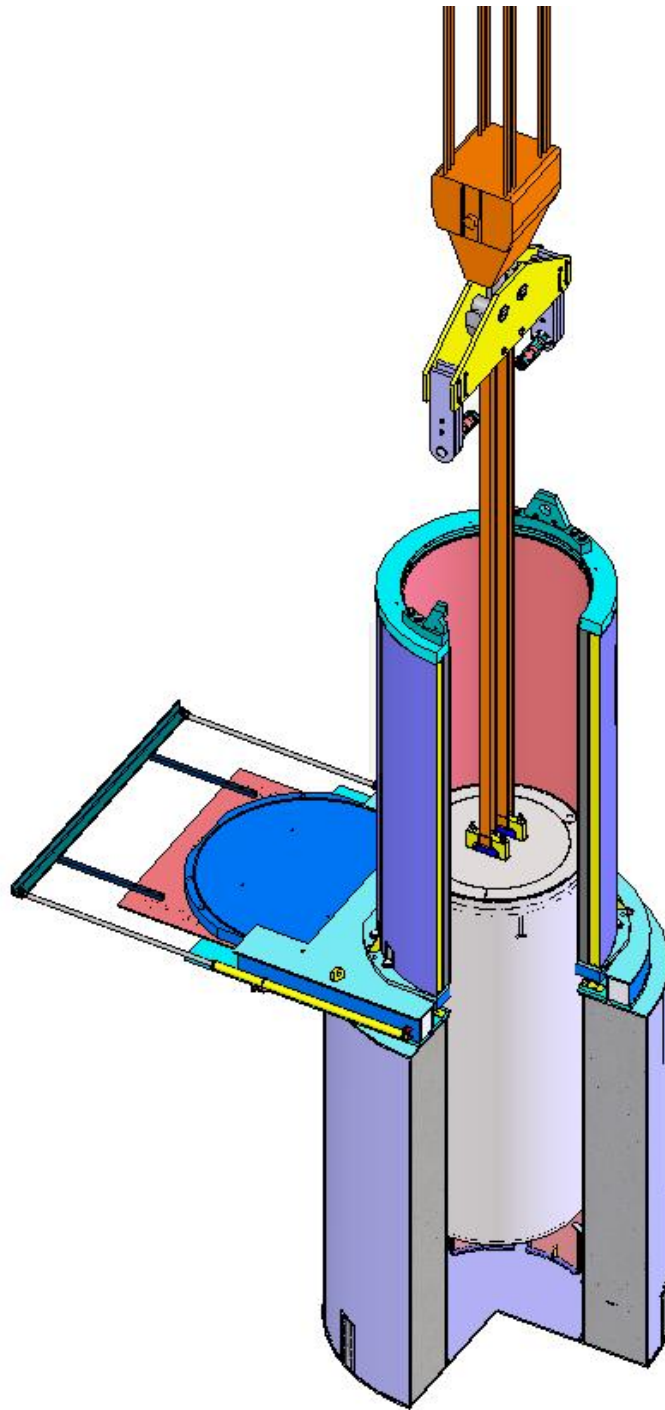
The principal ancillaries used in the site implementation of the HI-STORM FW System are summarized in Section 1.2 and referenced in Chapter 9 in the context of loading operations. A listing of common ancillaries needed by the host site is provided in Table 9.2.1. The detailed design of these ancillaries is not specified in this FSAR. In some cases, there are multiple distinct ancillary designs available for a particular application (such as a forced helium dehydrator or a vacuum drying system for drying the MPC) and as such, not every ancillary will be needed by every site. Ancillary designs are typically specific to a site to meet ALARA and personnel safety objectives.



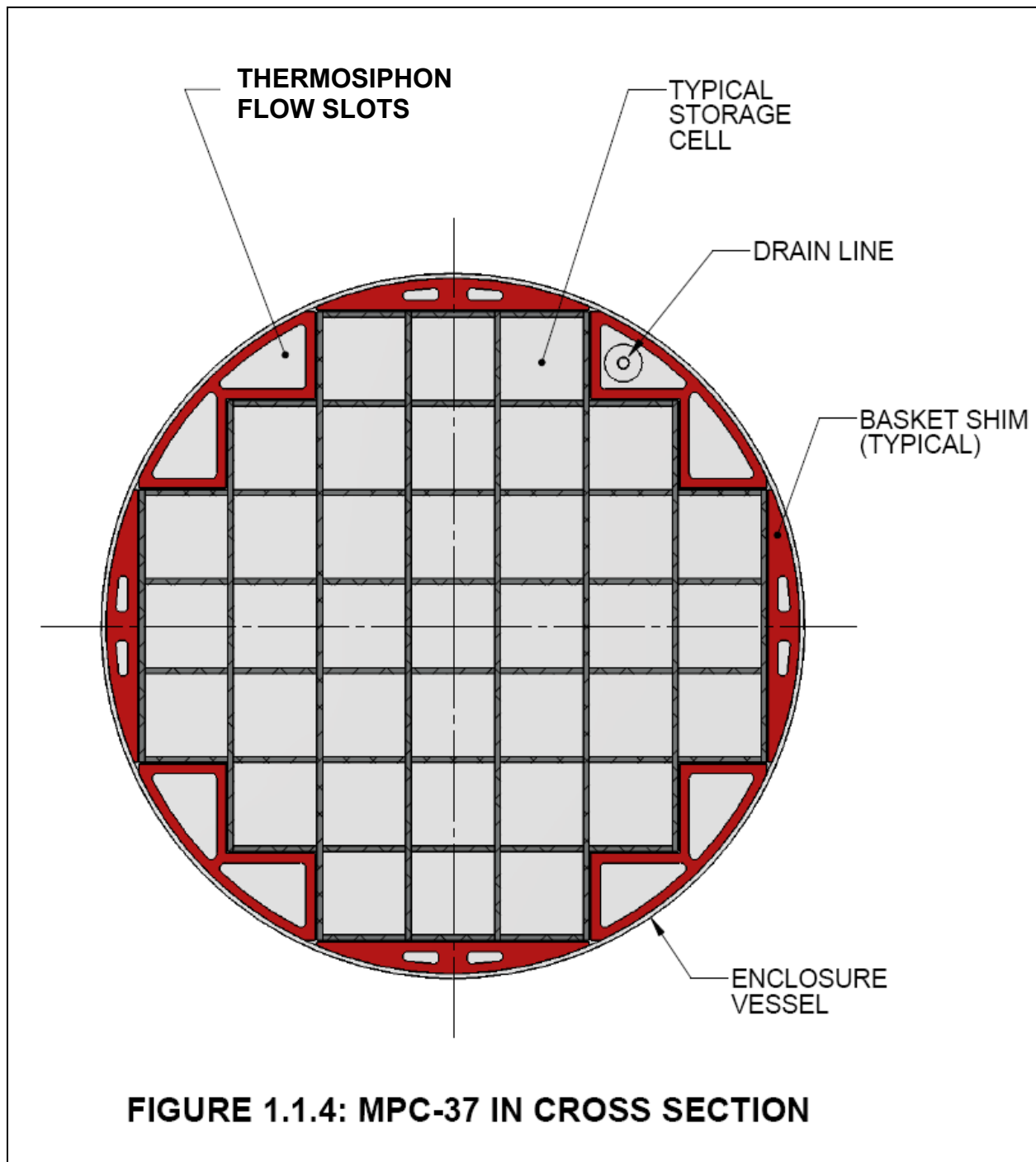
**FIGURE 1.1.1: HI-STORM FW VVM AND MPC-37 IN CUTAWAY VIEW**

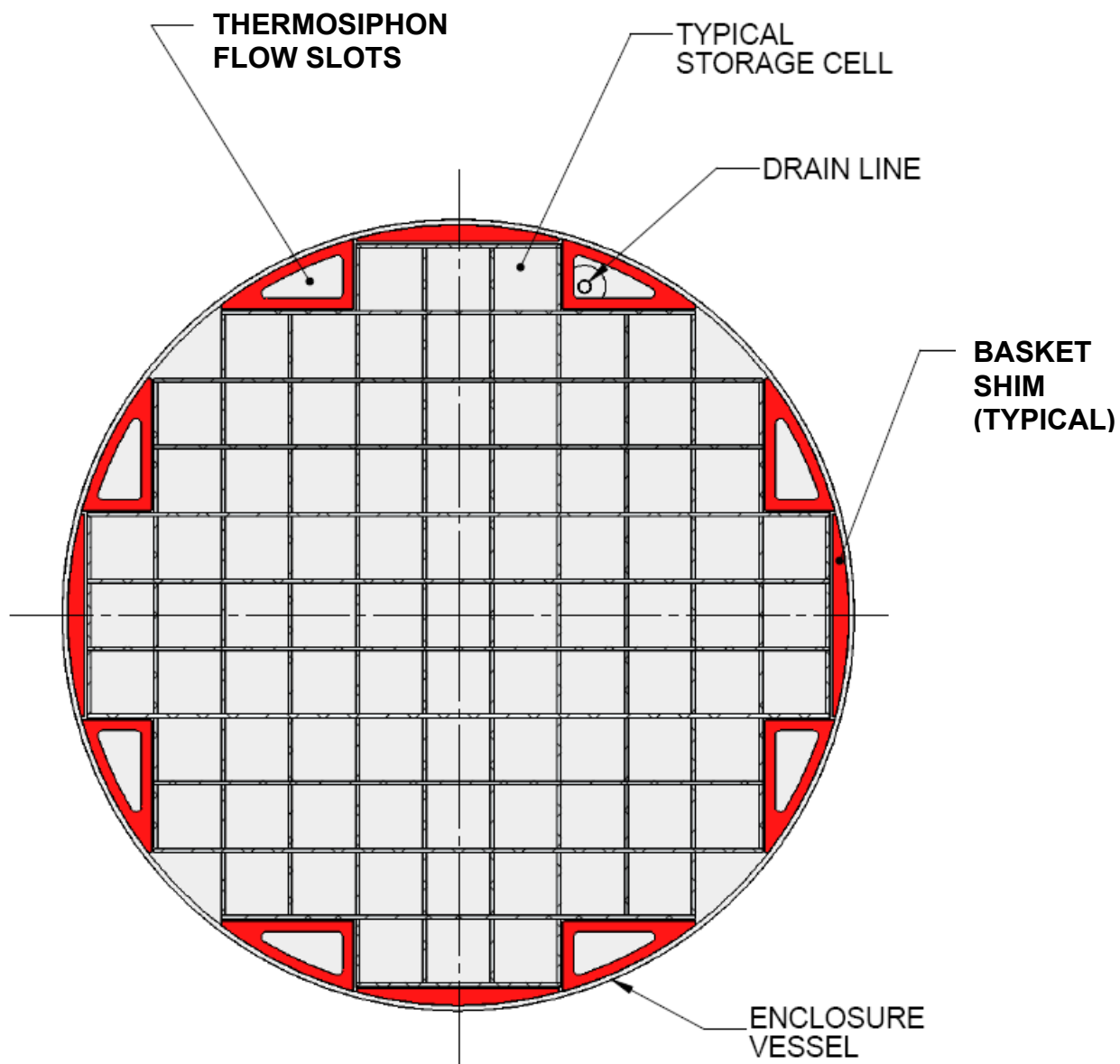


**FIGURE 1.1.2: MPC TRANSFER AT THE CANISTER TRANSFER FACILITY (PIT)**

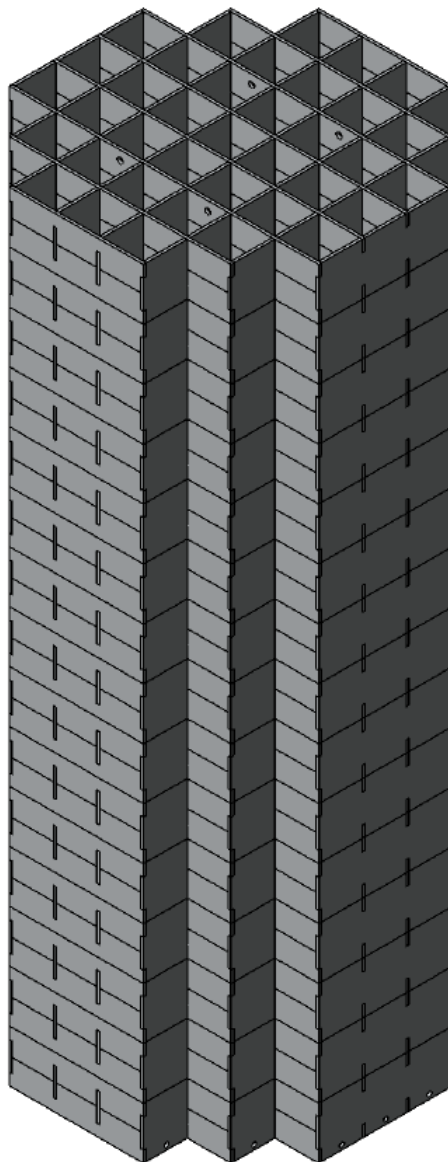


**FIGURE 1.1.3: MPC TRANSFER IN THE PLANT'S EGRESS BAY**



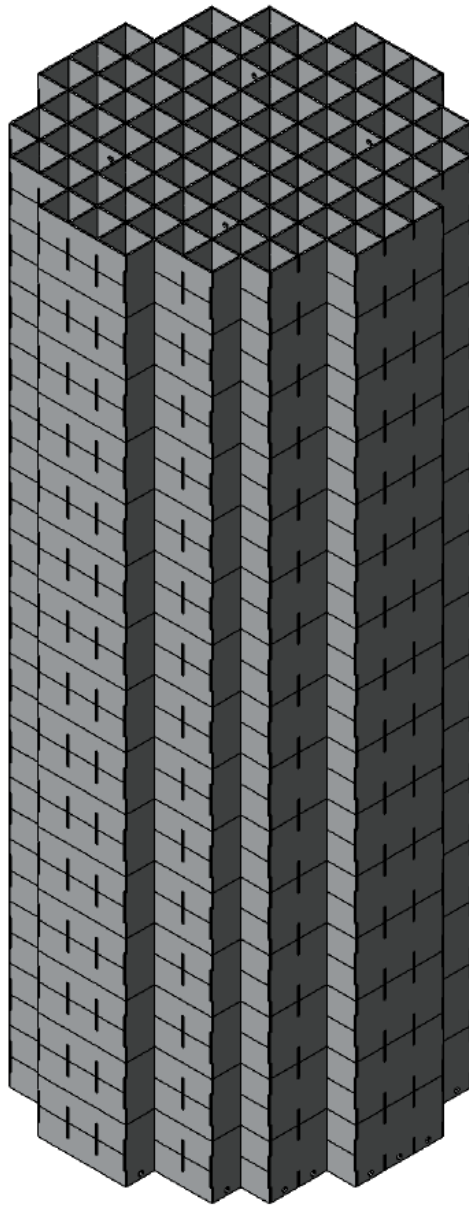


**FIGURE 1.1.5: MPC-89 IN CROSS SECTION**

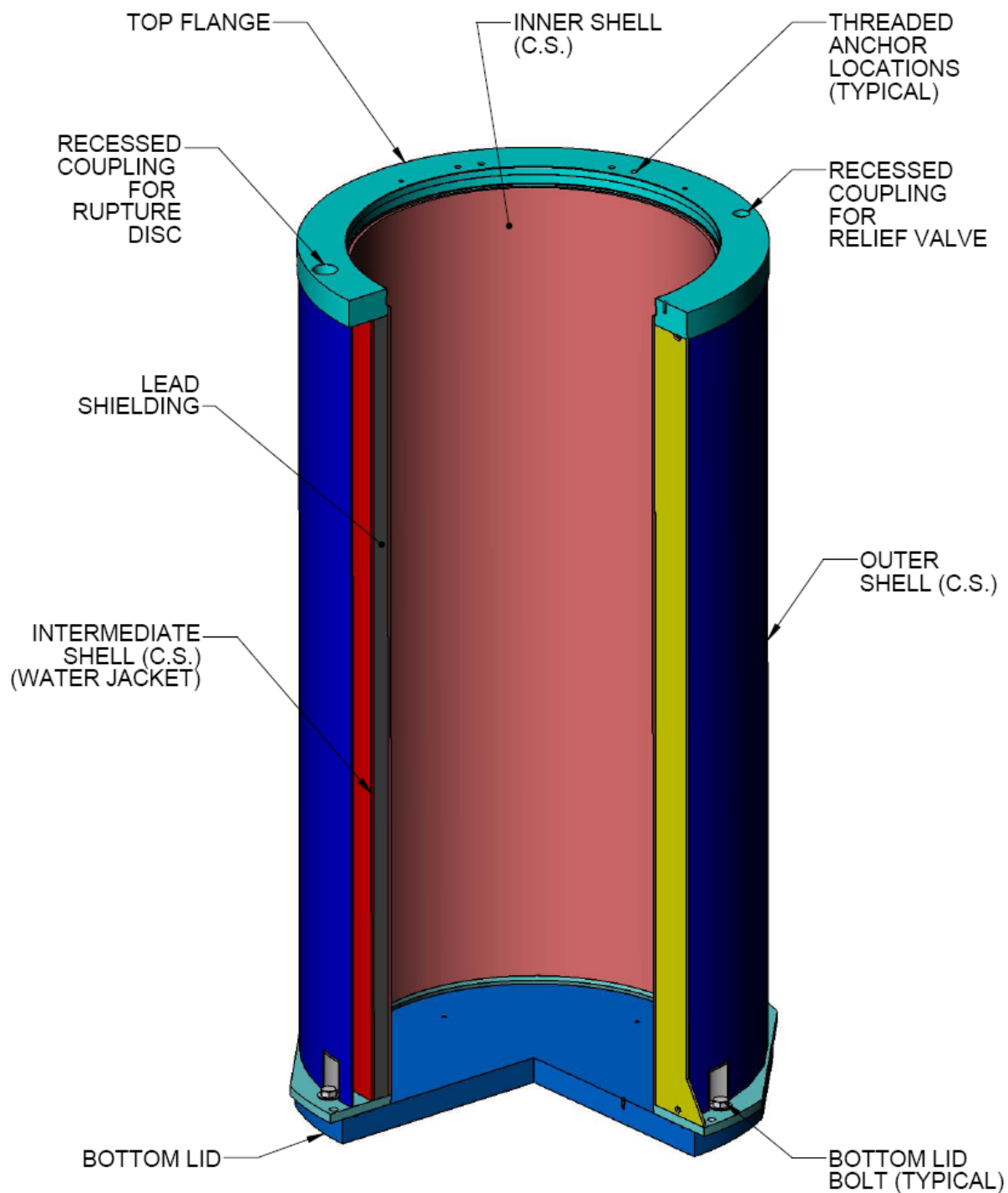


**FIGURE 1.1.6: PWR FUEL BASKET (37 STORAGE CELLS) IN PERSPECTIVE VIEW**





**FIGURE 1.1.7: BWR FUEL BASKET (89 STORAGE CELLS) IN PERSPECTIVE VIEW**



**FIGURE 1.1.8: CUTAWAY VIEW OF HI-TRAC VW**

## 1.2 GENERAL DESCRIPTION OF HI-STORM FW SYSTEM

### 1.2.1 System Characteristics

The HI-STORM FW System consists of interchangeable MPCs, which maintain the configuration of the fuel and is the confinement boundary between the stored spent nuclear fuel and the environment; and a storage overpack that provides structural protection and radiation shielding during long-term storage of the MPC. In addition, a transfer cask that provides the structural and radiation protection of an MPC during its loading, unloading, and transfer to the storage overpack is also subject to certification by the USNRC. Figure 1.1.1 provides a cross sectional view of the HI-STORM FW System with an MPC inserted into HI-STORM FW. Both casks (storage overpack and transfer cask) and the MPC are described below. The description includes information on the design details significant to their functional performance, fabrication techniques and safety features. All structures, systems, and components of the HI-STORM FW System, which are identified as Important-to-Safety (ITS), are specified on the licensing drawings provided in Section 1.5.

There are three types of components subject to certification in the HI-STORM FW docket (see Table 1.0.1).

- i. The multi-purpose canister (MPC)
- ii. The storage overpack (HI-STORM)
- iii. The transfer cask (HI-TRAC)

A listing of the common ancillaries not subject to certification but which may be needed by the host site to implement this system is provided in Table 9.2.1.

To ensure compatibility with the HI-STORM FW overpack, MPCs have identical external diameters. Due to the differing storage contents of each MPC, the loaded weight differs among MPCs (see Table 3.2.4 for loaded MPC weight data). Tables 1.2.1 and 1.2.2 contain the key system data and parameters for the MPCs.

The HI-STORM FW System shares certain common attributes with the HI-STORM 100 System, Docket No. 72-1014, namely:

- i. the honeycomb design of the MPC fuel basket;
- ii. the effective distribution of neutron and gamma shielding materials within the system;
- iii. the high heat dissipation capability;
- iv. the engineered features to promote convective heat transfer by passive means;
- v. a structurally robust steel-concrete-steel overpack construction.

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

The composite shell construction in the overpack, steel-concrete-steel, allows ease of fabrication and eliminates the need for the sole reliance on the strength of concrete.

A description of each of the components is provided in this section, along with fabrication and safety feature information.

#### **1.2.1.1 Multi-Purpose Canisters**

The MPC enclosure vessels are cylindrical weldments with identical and fixed outside diameters. Each MPC is an assembly consisting of a honeycomb fuel basket (Figures 1.1.6 and 1.1.7), a baseplate, a canister shell, a lid, and a closure ring. The number of SNF storage locations in an MPC depends on the type of fuel assembly (PWR or BWR) to be stored in it.

Subsection 1.2.3 and Table 1.2.1 summarize the allowable contents for each MPC model listed in Table 1.0.1. Subsection 2.1.8 provides the detailed specifications for the contents authorized for storage in the HI-STORM FW System. Drawings for the MPCs are provided in Section 1.5.

The MPC enclosure vessel is a fully welded enclosure, which provides the confinement for the stored fuel and radioactive material. The MPC baseplate and shell are made of stainless steel (Alloy X, see Appendix 1.A). The lid is a two piece construction, with the top structural portion made of Alloy X. The confinement boundary is defined by the MPC baseplate, shell, lid, port covers, and closure ring.

The HI-STORM FW System MPCs shares external and internal features with the HI-STORM 100 MPCs certified in the §72-1014 docket, as summarized below.

- i. MPC-37 and MPC-89 have an identical enclosure vessel which mimics the enclosure vessel design details used in the HI-STORM 100 counterparts including the shell thickness, the vent and drain port sizes, construction details of the top lid and closure ring, and closure weld details. The baseplate is made slightly thicker to ensure its bending rigidity is

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comparable to its counterpart in the HI-STORM 100 system. The material of construction of the pressure retaining components is also identical (options of austenitic stainless steels, denoted as Alloy X, is explained in Appendix 1.A herein as derived from the HI-STORM 100 FSAR with appropriate ASME Code edition updates). There are no gasketed joints in the MPCs.

- ii. The top lid of the MPCs contains the same attachment provisions for lifting and handling the loaded canister as the HI-STORM 100 counterparts.
- iii. The drain pipe and sump in the bottom baseplate of the MPCs (from which the drain pipe extracts the water during the dewatering operation) are also similar to those in the HI-STORM 100 counterparts.
- iv. The fuel basket is assembled from a rectilinear gridwork of plates so that there are no bends or radii at the cell corners. This structural feature eliminates the source of severe bending stresses in the basket structure by eliminating the offset between the cell walls which transfer the inertia load of the stored SNF to the basket/MPC interface during the various postulated accident events (such as non-mechanistic tipover). This structural feature is shared with the HI-STORM 100 counterparts. Figures 1.1.6 and 1.1.7 show the PWR and BWR fuel baskets, respectively, in perspective view.
- v. Precision extruded and/or machined blocks of aluminum alloy with axial holes (basket shims) are installed in the peripheral space between the fuel basket and the enclosure vessel to provide conformal contact surfaces between the basket shims and the fuel basket and between the basket shims and the enclosure vessel shell. The axial holes in the basket shims serve as the passageway for the downward flow of the helium gas under the thermosiphon action. This thermosiphon action is common to all MPCs including those of the HI-STORM 100. Various options are available to install these extruded shims in the basket periphery as summarized in Table 1.2.9.
- vi. To facilitate an effective convective circulation inside the MPC, the operating pressure is set the same as that in the HI-STORM 100 counterparts.
- vii. Like the high capacity baskets in the HI-STORM 100 MPCs, the fuel baskets do not contain flux traps.

Because of the above commonalities, the HI-STORM FW System is loaded in the same manner as the HI-STORM 100 system, and will use similar ancillary equipment, (e.g., lift attachments, lift yokes, lid welding machine, weld removal machine, cask transporter, mating device, low profile transporter or zero profile transporter, drying system, the hydrostatic pressure test system).

Lifting lugs, attached to the inside surface of the MPC shell, are used to place the empty MPC into the HI-TRAC VW transfer cask. The lifting lugs also serve to axially locate the MPC lid prior to welding. These internal lifting lugs cannot be used to handle a loaded MPC. The MPC lid is installed prior to any handling of a loaded MPC and there is no access to the internal lifting lugs once the MPC lid is installed.

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The MPC incorporates a redundant closure system. The MPC lid is edge-welded (welds are depicted in the licensing drawing in Section 1.5) to the MPC outer shell. The lid is equipped with vent and drain ports that are utilized to remove moisture from the MPC and backfill the MPC with a specified amount of inert gas (helium). The vent and drain ports are closed tight and covered with a port cover (plate) that is seal welded before the closure ring is installed. The closure ring is a circular ring edge-welded to the MPC shell and lid; it covers the MPC lid-to-shell weld and the vent and drain port cover plates. The MPC lid provides sufficient rigidity to allow the entire MPC loaded with SNF to be lifted by the suitably sized threaded anchor locations (TALs) in the MPC lid.

As discussed later in this section, the height of the MPC cavity plays a direct role in setting the amount of shielding available in the transfer cask. To maximize shielding and achieve ALARA within the constraints of a nuclear plant (such as crane capacity), it is necessary to minimize the cavity height of the MPC to the length of the fuel to be stored in it. Accordingly, the height of the MPC cavity is customized for each fuel type listed in Section 2.1. Table 3.2.1 provides the data to set the MPC cavity length as a small adder to the nominal fuel length (with any applicable NFH) to account for manufacturing tolerance, irradiation growth and thermal expansion effects.

For fuel assemblies that are shorter than the MPC cavity length (such as those without a control element in PWR SNF) a fuel shim may be utilized (as appropriate) to reduce the axial gap between the fuel assembly and the MPC cavity to approximately 1.5-2.5 inches. A small axial clearance is provided to account for manufacturing tolerances and the irradiation and thermal growth of the fuel assemblies. The actual length of fuel shims (if required) will be determined on a site-specific and fuel assembly-specific basis.

All components of the MPC assembly that may come into contact with spent fuel pool water or the ambient environment are made from stainless steel alloy or aluminum/aluminum alloy materials. Prominent among the aluminum based materials used in the MPC is the Metamic-HT neutron absorber lattice that comprises the fuel basket. As discussed in Chapter 8, concerns regarding interaction of coated carbon steel materials and various MPC operating environments [1.2.1] are not applicable to the HI-STORM FW MPCs. All structural components in an MPC enclosure vessel shall be made of Alloy X, a designation whose origin, as explained in the HI-STORM 100 FSAR [1.1.3], lies in the U.S. DOE's repository program.

As explained in Appendix 1.A, Alloy X (as defined in this FSAR) may be one of the following materials.

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

Any stainless steel part in an MPC may be fabricated from any of the acceptable Alloy X materials listed above.

The Alloy X group approach is accomplished by qualifying the MPC for all mechanical, structural, radiological, and thermal conditions using material thermo-physical properties that 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, a material has been defined that is referred to as Alloy X, whose thermo-physical properties, from the MPC design perspective, are the least favorable of the above four candidate materials.

The evaluation of the candidate Alloy X materials 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, it guarantees that the performance of the MPC will exceed the analytical predictions contained in this document.

The principal materials used in the manufacturing of the MPC are listed in the licensing drawings (Section 1.5) and the acceptance criteria are provided in Chapter 10. A listing of the fabrication specifications utilized in the manufacturing of HI-STORM FW System components is provided in Table 1.2.7. The specifications, procedures for sizing, forming machining, welding, inspecting, cleaning, and packaging of the completed equipment implemented by the manufacturer on the shop floor are required to conform to the fabrication specification in the above referenced tables.

#### **1.2.1.2 HI-STORM FW Overpack**

HI-STORM FW is a vertical ventilated module engineered to be fully compatible with the HI-TRAC VW transfer cask and the MPCs listed in Table 1.0.1. The HI-STORM FW overpack consists of two major parts:

- a. A dual wall cylindrical container with a set of inlet ducts near its bottom extremity and an integrally welded baseplate.
- b. A removable top lid equipped with a radially symmetric exit vent system.

The HI-STORM FW overpack is a rugged, heavy-walled cylindrical vessel. Figure 1.1.1 provides a pictorial view of the HI-STORM FW overpack with the MPC-37 partially inserted. The main structural function of the storage overpack is provided by carbon steel, and the main shielding function is provided by plain concrete. The overpack plain concrete is enclosed by a steel weldment of cylindrical shells, a thick baseplate, and a top annular plate. A set of four equally spaced radial connectors join the inner and outer shells and define a fixed width annular space for placement of concrete. The overpack lid also has concrete to provide neutron and gamma shielding.

The storage overpack provides an internal cylindrical cavity of sufficient height and diameter for housing an MPC (Figure 1.1.1) with an annular space between the MPC enclosure vessel and the overpack for ventilation air flow. The upward flowing air in the annular space (drawn from the ambient by a purely passive action), extracts heat from the MPC surface by convective heat transfer. The rate of air flow is governed by the amount of heat in the MPC (i.e., the greater the heat load, the greater the air flow rate).

To maximize the cooling action of the ventilation air stream, the ventilation flow path is optimized to minimize hydraulic resistance. The HI-STORM FW features eight inlet ducts. Each duct is narrow and tall and of an internally refractive contour which minimizes radiation streaming while optimizing the hydraulic resistance of airflow passages. The inlet air duct design, referred to as the “Radiation Absorbent Duct,” is subject to an ongoing action on a provisional Holtec International patent application by the USPTO (ca. March 2009) and is depicted in the licensing drawing in Section 1.5. The Radiation Absorbent Duct also permits the MPC to be placed directly on the baseplate of the overpack instead of on a pedestal that would raise it above the duct.

An array of radial tube-type gussets (MPC guides) welded to the inner shell and the baseplate are shaped to guide the MPC during MPC transfer and ensure it is centered within the overpack. The MPC guides have an insignificant effect on the overall hydraulic resistance of the ventilation air stream. Furthermore, the top array of MPC guides are longitudinally oriented members, sized and aligned to serve as impact attenuators which will crush against the solid MPC lid during an impactive collision, such as a non-mechanistic tip-over scenario.

The height of the storage cavity in the HI-STORM FW overpack is set equal to the height of the MPC plus a fixed amount to allow for thermal growth effects and to provide for adequate ventilation space (low hydraulic resistance) above the MPC (See Table 3.2.1).

The outlet duct is located in the overpack lid (Figure 1.1.1) pursuant to Holtec Patent No. 6,064,710. The outlet duct opening is narrow in height which reduces the radiation streaming path from the contents, however, aside from the minor interference from the support plates, the duct extends circumferentially 360° which significantly increases the flow area and in-turn minimizes hydraulic resistance.

The overpack lid, like the body, is also a steel weldment filled with plain concrete. The lid is equipped with a radial ring welded to its underside which provides additional shielding for the MPC/overpack annulus. The radial ring also serves to center the lid on the overpack body. A third, equally important function of the radial ring is to prevent the lid from sliding across the top surface of the overpack body during a non-mechanistic tip-over event.

Within the ducts, an array of duct photon attenuators (DPAs) may be installed (Holtec Patent No.6,519,307B1) to further decrease the amount of radiation scattered to the environment. These Duct Photo Attenuators (DPAs) are designed to scatter any radiation streaming through the ducts. Scattering the radiation in the ducts reduces the streaming through the overpack penetration resulting in a significant decrease in the local dose rates. The configuration of the DPAs is such that the increase in the resistance to flow in the air inlets and outlets is minimized. The DPAs are not credited in the safety analyses performed in this FSAR, nor are they depicted in the licensing drawings. DPAs can be used at a site if needed to lower site boundary dose rates with an appropriate site-specific engineering evaluation.

Each duct opening is equipped with a heavy duty insect barrier (screen). Routine inspection of the screens or temperature monitoring of the air exiting the outlet ducts is required to ensure that a blockage of the screens is detected and removed in a timely manner. The evaluation of the effects of partial and complete blockage of the air ducts is considered in Chapter 12 of this FSAR.

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Four threaded anchor blocks at the top of the overpack are provided for lifting. The anchor blocks are integrally welded to the radial plates which join the overpack inner and outer steel shells. The four anchor blocks are located at 90° angular spacing around the circumference of the top of the overpack body.

The internal surfaces of the HI-STORM FW overpack facing the MPC may be optionally equipped with a heat shield made of a thin steel sheet stock to limit the radiant heat delivered to the overpacks's body.

The plain concrete between the overpack inner and outer steel shells and the lid is specified to provide the necessary shielding properties (dry density) and compressive strength. The shielding concrete shall be in accordance with the requirements specified in Appendix 1.D of the HI-STORM 100 FSAR [1.1.3] and Table 1.2.5 herein. Commitment to follow the specification of plain concrete in the HI-STORM 100 FSAR in this docket ensures that a common set of concrete placement procedures will be used in both overpack types which will be important for configuration control at sites where both systems may be deployed.

The principal function of the concrete is to provide shielding against gamma and neutron radiation. However, the massive bulk of concrete imparts a large thermal inertia to the HI-STORM FW overpack, allowing it to moderate the rise in temperature of the system under hypothetical conditions when all ventilation passages are assumed to be blocked. During the postulated fire accident the high thermal inertia characteristics of the HI-STORM FW concrete control the temperature of the MPC. Although the annular concrete mass in the overpack shell is not a structural member, it does act as an elastic/plastic filler of the inter-shell space buttressing the steel shells.

Density and compressive strength are the key parameters that bear upon the performance of concrete in the HI-STORM FW System. For evaluating the physical properties of concrete for completing the analytical models, conservative formulations of Reference [1.2.2] are used.

Thermal analyses, presented in Chapter 4, show that the temperatures during normal storage conditions do not threaten the physical integrity of the HI-STORM FW overpack concrete.

The principal materials used in the manufacturing of the overpack are listed in the licensing drawings and the acceptance criteria are provided in Chapter 10. Tables 1.2.6 and 1.2.7 provide applicable code paragraphs for manufacturing the HI-STORM FW overpack.

### **1.2.1.3 HI-TRAC VW Transfer Cask**

The HI-TRAC VW transfer cask (Figure 1.1.8) is engineered to be used to perform all short-term loading operations on the MPC beginning with fuel loading and ending with the emplacement of the MPC in the storage overpack. The HI-TRAC VW is also used for short term unloading operations beginning with the removal of the MPC from the storage overpack and ending with fuel unloading.

HI-TRAC VW is designed to meet the following specific performance objectives that are centered on ALARA and physical safety of the plant's operations staff.

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- a. Provide maximum shielding to the plant personnel engaged in conducting short-term operations.
- b. Provide protection of the MPC against extreme environmental phenomena loads, such as tornado-borne missiles, during short-term operations.
- c. Serve as the container equipped with the appropriate lifting appurtenances in accordance with ANSI N14.6 [1.2.3] to lift, move, and handle the MPC, as required, to perform the short-term operations.
- d. Provide the means to restrain the MPC from sliding and protruding beyond the shielding envelope of the transfer cask under a (postulated) handling accident.
- e. Facilitate the transfer of a loaded MPC to or from the HI-STORM FW overpack (or another physically compatible storage or transfer cask) by vertical movement of the MPC without any risk of damage to the canister by friction.

The above performance demands on the HI-TRAC VW are met by its design configuration as summarized below and presented in the licensing drawings in Section 1.5.

HI-TRAC VW is principally made of carbon steel and lead. The cask consists of two major parts, namely (a) a multi-shell cylindrical cask body, and (b) a quick connect/disconnect bottom lid. The cylindrical cask body is made of three concentric shells joined to a solid annular top flange and a solid annular bottom flange by circumferential welds. The innermost and the middle shell are fixed in place by longitudinal ribs which serve as radial connectors between the two shells. The radial connectors provide a continuous path for radial heat transfer and render the dual shell configuration into a stiff beam under flexural loadings. The space between these two shells is occupied by lead, which provides the bulk of the transfer cask's gamma radiation shielding capability and accounts for a major portion of its weight.

Between the middle shell and the outermost shell is the weldment that is referred to as the "water jacket." The water jacket is filled with water and may contain ethylene glycol fortified water, if warranted by the environmental conditions at the time of use. The water jacket provides most of the neutron shielding capability to the cask. The water jacket is outfitted with pressure relief devices to prevent over-pressurization in the case of an off-normal or accident event that causes the water mass inside of it to boil.

The water in the water jacket serves as the neutron shield when required. When the cask is being removed from the pool and the MPC is full of water, the water jacket can be empty. This will minimize weight, if for example, crane capacities are limited, since the water within the MPC cavity is providing the neutron shielding during this time. However, the water jacket must be filled before the MPC is emptied of water. This keeps the load on the crane (i.e., weight of the loaded transfer cask) nearly constant between the lifts before and after MPC processing. Furthermore, the amount of shielding provided by the transfer cask is maximized at all times within crane capacity constraints. The water jacket concept is disclosed in a Holtec Patent [6,587,536 B1].

As the description of loading operations in Chapter 9 of this FSAR indicates, most of the human activities occur near the top of the transfer cask. Therefore, the geometry of the transfer cask is configured to maximize shielding by eliminating penetrations and discontinuities such as lifting trunnions. Instead, the HI-TRAC VW is lifted using a pair of lift blocks that are anchored into the

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top forging of the transfer cask using a set of high strength bolts. An optional device which prevents the MPC from sliding out of the transfer cask is attached to the lift blocks.

The bottom of the transfer cask is equipped with a thick lid. It is provided with a gasket seal against the machined face of the bottom flange creating a watertight (open top) container. A set of bolts that tap into the machined holes in the bottom lid provide the required physical strength to meet the structural imperatives of ANSI N14.6 and as well as bolt pull to maintain joint integrity. The bottom lid can be fastened and released from the cask body by accessing its bolts from above the transfer cask bottom flange, which is an essential design feature to permit MPC transfer operations described in Chapter 9.

To optimize the shielding in the body of HI-TRAC VW, two design strategies have been employed;

1. The height of the HI-TRAC's cavity is set to its optimal value (slightly greater than the MPC height as specified in Table 3.2.1), therefore allowing more shielding to be placed in the radial direction of the transfer cask.
2. The thickness of the lead in the transfer cask shall be customized for the host site. The thickness of the lead cylinder can be varied within the limits given in Table 3.2.2. The nominal radial thickness of the water jacket is fixed and therefore the outside diameter of the HI-TRAC will vary accordingly.

The above design approach permits the quantity of shielding around the body of the transfer cask to be maximized for a given length and weight of fuel in keeping with the practices of ALARA. At some host sites, a lead thickness greater than allowed by Table 3.2.2 may be desirable and may be feasible but will require a site-specific safety evaluation.

The use of the suffix VW in the HI-TRAC's designation is intended to convey this **Variable Weight** feature incorporated by changing the HI-TRAC height and lead thickness to best accord with the MPC height and plant's architecture. Table 3.2.6 provides the operating weight data for a HI-TRAC VW when handling the Reference PWR and BWR fuel in Table 1.0.4.

The principal materials used in the manufacturing of the transfer cask are listed in the licensing drawings and the acceptance criteria are provided in Chapter 10. Tables 1.2.6 and 1.2.7 provide applicable code paragraphs for manufacturing the HI-TRAC VW.

#### **1.2.1.4 Shielding Materials**

Steel and concrete are the principal shielding materials in the HI-STORM FW overpack. The steel and concrete shielding materials in the lid provide additional gamma attenuation to reduce both direct and skyshine radiation. The combination of these shielding materials ensures that the radiation and exposure objectives of 10CFR72.104 and 10CFR72.106 are met.

Steel, lead, and water are the principal shielding materials in the HI-TRAC transfer cask. The combination of these three shielding materials ensures that the radiation and exposure objectives of 10CFR72.106 and ALARA are met. The extent and location of shielding in the transfer cask plays

an important role in minimizing the personnel doses during loading, handling, and transfer.

The MPC fuel basket structure provides the initial attenuation of gamma and neutron radiation emitted by the radioactive contents. The MPC shell, baseplate, and thick lid provide additional gamma attenuation to reduce direct radiation.

#### 1.2.1.4.1 Neutron Absorber – Metamic HT

Metamic-HT is the designated neutron absorber in the HI-STORM FW MPC baskets. It is also the structural material of the basket. The properties of Metamic-HT and key characteristics, necessary for ensuring nuclear reactivity control, thermal, and structural performance of the basket, are presented below.

**[Withheld in Accordance with 10 CFR 2.390]**

#### 1.2.1.4.2 Neutron Shielding

Neutron shielding in the HI-STORM FW overpack is provided by the thick walls of concrete contained inside the steel vessel and the top lid. Concrete is a shielding material with a long proven history in the nuclear industry. The concrete composition has been specified to ensure its continued integrity under long term temperatures required for SNF storage.

The specification of the HI-STORM FW overpack neutron shielding material is predicated on functional performance criteria. These criteria are:

- Attenuation of neutron radiation to appropriate levels;
- Durability of the shielding material under normal conditions (i.e. under normal condition 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. Final specification of a shield material is a result of optimizing the material properties with respect to the above criteria, along with the design of the shield system, to achieve the desired shielding results.

The HI-TRAC VW transfer cask is equipped with a water jacket providing radial neutron shielding. The water in the water jacket may be fortified with ethylene glycol to prevent freezing under low

temperature operations [1.2.4].

During certain evolutions in the short term handling operations, the MPC may contain water which will supplement neutron shielding.

#### 1.2.1.4.3 Gamma Shielding Material

Gamma shielding in the HI-STORM FW storage overpack is primarily provided by massive concrete sections contained in the robust steel vessel. The carbon steel in the overpack supplements the concrete gamma shielding. To reduce the radiation streaming through the overpack penetrations, duct photon attenuators may be installed (as discussed previously in section 1.2.1.2) to further decrease radiation streaming from the ducts.

In the HI-TRAC VW transfer cask, the primary gamma shielding is provided by lead. As in the storage overpack, carbon steel supplements the lead gamma shielding of the HI-TRAC VW transfer cask.

In the MPC, the gamma shielding is provided by its stainless steel enclosure vessel (including a thick lid); and its aluminum based fuel basket and aluminum alloy basket shims.

#### 1.2.1.5 **Lifting Devices**

Lifting and handling of the loaded HI-STORM FW overpack is carried out in the vertical upright configuration using the threaded anchor blocks arranged circumferentially at 90° spacing around the overpack. These anchor blocks are used for overpack lifting as well as securing the overpack lid to the overpack body. The storage overpack may be lifted with a lifting device that engages the anchor blocks with threaded studs and connects to a crane or similar equipment. The overpack anchor blocks are integral to the overpack and designed in accordance with Regulatory Guide 3.61. All lifting appurtenances used with the HI-STORM FW overpack are designed in accordance with NUREG-0612 and ANSI N14.6, as applicable.

Like the storage overpack, the loaded transfer cask is also lifted using a specially engineered appurtenance denoted as the lift block in Table 9.1.2 and Figure 9.2.1. The top flange of the transfer cask is equipped with threaded holes that allow lifting of the loaded HI-TRAC in the vertical upright configuration. These threaded lifting holes are integral to the transfer cask and are designed in accordance with NUREG 0612. All lifting appurtenances used with the HI-TRAC VW are designed in accordance with NUREG-0612 and ANSI N14.6, as applicable.

The top of the MPC lid is equipped with eight threaded holes that allow lifting of the loaded MPC. These holes allow the loaded MPC to be raised and/or lowered through the HI-TRAC VW transfer cask using lifting attachments (functional equivalent of the lift blocks used with HI-TRAC VW). The threaded holes in the MPC lid are integral to the MPC and designed in accordance with NUREG 0612. All lifting appurtenances used with the MPC are designed in accordance with NUREG-0612 and ANSI N14.6, as applicable.

The transporter used to handle the loaded transfer cask or overpack during transport operations must

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be engineered to provide a high integrity handling of the load, defined as a lifting/handling operation wherein the risk of an uncontrolled lowering of the heavy load is non-credible. In handling equipment such as a transporter, high integrity handling is achieved through (a) a body and any vertical columns designed to comply with stress limits of ASME Section III, Subsection NF, Class 3, (b) an overhead beam that is single-failure-proof, and (c) redundant drop protection features. Single failure proof handling capability is achieved by ensuring that the applicable factor of safety is 200% of that required by the reference design code or national consensus standard. It is acceptable to have certain load carrying members (such as the lifting towers in a vertical cask transporter) designed with redundant devices and others (such as the transverse beam) designed to the doubled factor of safety in order to meet the criteria set above. Heavy load handling device criteria is detailed in TM-141 [Ref 1.2.15].

#### **1.2.1.6 Design Life**

The design life of the HI-STORM FW System is 60 years. 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 (see Chapter 8). A maintenance program, as specified in Chapter 10, is also implemented to ensure the service life of the HI-STORM FW System will exceed its design life of 60 years. The design considerations that assure the HI-STORM FW System performs as designed include the following:

##### HI-STORM FW Overpack and HI-TRAC VW Transfer Cask

- Exposure to Environmental Effects
- Material Degradation
- Maintenance and Inspection Provisions

##### MPCs

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

The adequacy of the HI-STORM FW System materials for its design life is discussed in Chapter 8. Transportability considerations pursuant to 10CFR72.236(m) are discussed in Section 2.4.

### **1.2.2 Operational Characteristics**

#### **1.2.2.1 Design Features**

The design features of the HI-STORM FW System, described in Subsection 1.2.1 in the foregoing, are intended to meet the following principal performance characteristics under all credible modes of operation:

- (a) Maintain subcriticality
- (b) Prevent unacceptable release of contained radioactive material
- (c) Minimize occupational and site boundary dose
- (d) Permit retrievability of contents (fuel must be retrievable from the MPC under normal and off-normal conditions in accordance with ISG-2 and the MPC must be recoverable after accident conditions in accordance with ISG-3)

Chapter 11 identifies the many design features built into the HI-STORM FW System to minimize dose and maximize personnel safety. Among the design features intrinsic to the system that facilitate meeting the above objectives are:

- i. The loaded HI-STORM FW overpack and loaded HI-TRAC VW transfer cask are always maintained in a vertical orientation during handling (with the rare exception of the transfer cask as described in Subsection 4.5.1).
- ii. The height of the HI-STORM FW overpack and HI-TRAC VW transfer cask is minimized consistent with the length of the SNF. This eliminates the need for major structural modifications at the plant and/or eliminates operational steps that impact ALARA.
- iii. The extent of shielding in the transfer cask is maximized at each plant within the crane and architectural limitations of the plant by minimizing the height in accordance with the length of the SNF to permit additional shielding material in the walls of the transfer cask.
- iv. The increased number of inlet ducts and the circumferential outlet vents in HI-STORM FW overpack are configured to make the thermal performance less susceptible to wind.
- v. Tall and narrow inlet ducts in the HI-STORM FW overpack in conjunction with the thermosiphon action in the MPC design, render the HI-STORM FW System more resistant to a thermally adverse flood condition (Section 2.2).
- vi. The design of the HI-STORM FW affords the user the flexibility to utilize higher density concrete than the minimum prescribed value in Table 1.2.5 to further reduce the site boundary dose.

The HI-STORM FW overpack utilizes the same cross-connected dual steel shell configuration used in other HI-STORM models. The dual shell steel weldment with an integrally connected baseplate forms a well defined annulus wherein plain concrete of the desired density is installed. While both steel and concrete in the overpack body are effective in neutron and gamma shielding, the principal role of the radially conjoined steel shell is to provide the structural rigidity to support the mass of the shielding concrete. As calculations in Chapter 3 show, the dual steel shell structure can support the mass of concrete of any available density with ample margin of safety. Consequently, the mass of concrete utilized to shield against the stored fuel is only limited by the density of the available

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aggregate. Users of HI-STORM 100 systems have used concrete of density approaching 200 lb/ft<sup>3</sup> to realize large dose reductions at ISFSIs to support site specific considerations.

The above comment also applies to the HI-STORM FW overpack lid, which is a massive steel weldment made of plate and shell segments filled with shielding concrete. The steel in the lid, while contributing principally to gamma shielding, provides the needed structural capacity. Concrete performs as a missile barrier and is critical to minimizing skyshine. High density concrete can also be used in the HI-STORM FW overpack lid if reducing skyshine is a design objective at a plant.

The site boundary dose from the HI-STORM FW System is minimized by using specially shaped ducts at the bottom of the overpack and in the lid. The ducts and the annular space between the stored MPC and the HI-STORM FW cavity serve to promote ventilation of air to reject the MPC's decay heat to the environment.

The criticality control features of the HI-STORM FW 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**

A summary sequence of loading operations necessary to defuel a spent fuel pool using the HI-STORM FW System (shown with MPC Transfer in the plant's Egress Bay) is shown in a series of diagrams in Figure 1.2.3. The loading sequence underscores the inherent simplicity of the loading evolutions and its compliance with ALARA. A more detailed sequence of steps for loading and handling operations is provided in Chapter 9, aided by illustrative figures, to serve as the guidance document for preparing site-specific implementation 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 absorbing materials in the fuel basket. The entire basket is made of Metamic-HT, a uniform dispersoid of boron carbide and nano-particles of alumina in an aluminum matrix, serves as the neutron absorber. This accrues four major safety and reliability advantages:

- (i) The larger B-10 areal density in the Metamic-HT allows higher enriched fuel (i.e., BWR fuel with planar average initial enrichments greater than 4.5 wt% U-235) without relying on gadolinium or burn-up credit.
- (ii) The neutron absorber cannot be removed from the basket or displaced within it.
- (iii) Axial movement of the fuel with respect to the basket has no reactivity consequence because the entire length of the basket contains the B-10 isotope.
- (iv) The larger B-10 areal density in the Metamic-HT reduces the reliance on soluble boron credit during loading/unloading of PWR fuel.

#### 1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the HI-STORM FW System. A detailed evaluation is provided in Section 3.4.

#### 1.2.2.3.3 Operation Shutdown Modes

The HI-STORM FW System is totally passive and consequently, operation shutdown modes are unnecessary.

#### 1.2.2.3.4 Instrumentation

As stated earlier, the HI-STORM FW MPC, which is seal welded, non-destructively examined, and pressure tested, confines the radioactive contents. The HI-STORM FW 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. At the option of the user, temperature elements may be utilized to monitor the air temperature of the HI-STORM FW overpack exit vents in lieu of routinely inspecting the vents for blockage.

#### 1.2.2.3.5 Maintenance Technique

Because of its passive nature, the HI-STORM FW System requires minimal maintenance over its lifetime. No special maintenance program is required. Chapter 10 describes the maintenance program set forth for the HI-STORM FW System.

### 1.2.3 Cask Contents

This sub-section contains information on the cask contents pursuant to 10 CFR72, paragraphs 72.2(a)(1),(b) and 72.236(a),(c),(h),(m).

The HI-STORM FW System is designed to house both BWR and PWR spent nuclear fuel assemblies. Tables 1.2.1 and 1.2.2 provide key system data and parameters for the MPCs. A description of acceptable fuel assemblies for storage in the MPCs is provided in Section 2.1. This includes fuel assemblies classified as damaged fuel assemblies and fuel debris in accordance with the definitions of these terms in the Glossary. All fuel assemblies, non-fuel hardware, and neutron sources authorized for packaging in the MPCs must meet the fuel specifications provided in Section 2.1. All fuel assemblies classified as damaged fuel or fuel debris must be stored in damaged fuel containers (DFC).

As shown in Figure 1.2.1 (MPC-37) and Figure 1.2.2 (MPC-89), each storage location is assigned to one of three regions, denoted as Region 1, Region 2, and Region 3 with an associated cell identification number. For example, cell identified as 2-4 is Cell 4 in Region 2. A DFC can be stored in the outer peripheral locations of both MPC-37 and MPC-89 as shown in Figures 2.1.1 and 2.1.2, respectively. The permissible heat loads for each cell, region, and the total canister are given in Tables 1.2.3 and 1.2.4 for MPC-37 and MPC-89, respectively. The sub-design heat loads for each cell, region and total canister are in Table 4.4.11.

TABLE 1.2.1		
KEY SYSTEM DATA FOR HI-STORM FW SYSTEM		
ITEM	QUANTITY	NOTES
Types of MPCs	2	1 for PWR 1 for BWR
MPC storage capacity <sup>†</sup> :	MPC-37	Up to 37 undamaged ZR clad PWR fuel assemblies with or without non-fuel hardware. Up to 12 damaged fuel containers containing PWR damaged fuel and/or fuel debris may be stored in the locations denoted in Figure 2.1.1 with the remaining basket cells containing undamaged fuel assemblies, up to a total of 37.
MPC storage capacity <sup>†</sup> :	MPC-89	Up to 89 undamaged ZR clad BWR fuel assemblies. Up to 16 damaged fuel containers containing BWR damaged fuel and/or fuel debris may be stored in locations denoted in Figure 2.1.2 with the remaining basket cells containing undamaged fuel assemblies, up to a total of 89.

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<sup>†</sup> See Chapter 2 for a complete description of authorized cask contents and fuel specifications.

TABLE 1.2.2		
KEY PARAMETERS FOR HI-STORM FW MULTI-PURPOSE CANISTERS		
Parameter	PWR	BWR
Pre-disposal service life (years)	100	100
Design temperature, max./min. (°F)	752 <sup>†</sup> /-40 <sup>††</sup>	752 <sup>†</sup> /-40 <sup>††</sup>
Design internal pressure (psig)		
Normal conditions	100	100
Off-normal conditions	120	120
Accident Conditions	200	200
Total heat load, max. (kW)	See Table 1.2.3	See Table 1.2.4
Maximum permissible peak fuel cladding temperature:		
Long Term Normal (°F)	752	752
Short Term Operations (°F)	752 or 1058 <sup>†††</sup>	752 or 1058 <sup>†††</sup>
Off-normal and Accident (°F)	1058	1058
Maximum permissible multiplication factor ( $k_{eff}$ ) including all uncertainties and biases	< 0.95	< 0.95
B <sub>4</sub> C content (by weight) (min.) in the Metamic-HT Neutron Absorber (storage cell walls)	10%	10%
<b>[Withheld in Accordance with 10 CFR 2.390]</b>	<b>[Withheld in Accordance with 10 CFR 2.390]</b>	<b>[Withheld in Accordance with 10 CFR 2.390]</b>
<b>[Withheld in Accordance with 10 CFR 2.390]</b>	<b>[Withheld in Accordance with 10 CFR 2.390]</b>	<b>[Withheld in Accordance with 10 CFR 2.390]</b>
End closure(s)	Welded	Welded
Fuel handling	Basket cell openings compatible with standard grapples	Basket cell openings compatible with standard grapples
Heat dissipation	Passive	Passive

<sup>†</sup> Maximum normal condition design temperatures for the MPC fuel basket. A complete listing of design temperatures for all components is provided in Table 2.2.3.

<sup>††</sup> Temperature based on off-normal minimum environmental temperatures specified in Section 2.2.2 and no fuel decay heat load.

<sup>†††</sup> See Section 4.5 for discussion of the applicability of the 1058°F temperature limit during short-term operations, including MPC drying.

TABLE 1.2.3 MPC-37 HEAT LOAD DATA (See Figure 1.2.1)					
Number of Regions: 3					
Number of Storage Cells: 37					
Maximum Design Basis Heat Load (kW): 44.09 (Pattern A); 45.0 (Pattern B)					
Region No.	Decay Heat Limit per Cell, kW		Number of Cells per Region	Decay Heat Limit per Region, kW	
	Pattern A	Pattern B		Pattern A	Pattern B
1	1.05	1.0	9	9.45	9.0
2	1.70	1.2	12	20.4	14.4
3	0.89	1.35	16	14.24	21.6

Note: See Chapter 4 for decay heat limits per cell when vacuum drying high burnup fuel.

TABLE 1.2.4  
MPC-89 HEAT LOAD DATA (See Figure 1.2.2)

Number of Regions: 3			
Number of Storage Cells: 89			
Maximum Design Basis Heat Load: 46.36 kW			
Region No.	Decay Heat Limit per Cell, kW	Number of Cells per Region	Decay Heat Limit per Region, kW
1	0.44	9	3.96
2	0.62	40	24.80
3	0.44	40	17.60

Note: See Chapter 4 for decay heat limits per cell when loading high burnup fuel and using vacuum drying of the MPC.

TABLE 1.2.5  
CRITICALITY AND SHIELDING SIGNIFICANT SYSTEM DATA

<b>Item</b>	<b>Property</b>	<b>Value</b>
Metamic-HT Neutron Absorber	Nominal Thickness (mm)	10 (MPC-89) 15 (MPC-37)
	Minimum B <sub>4</sub> C Weight %	10 (MPC-89) 10 (MPC-37)
Concrete in HI-STORM FW overpack body and lid	Installed Nominal Density (lb/ft <sup>3</sup> )	150 (reference) 200 (maximum)

TABLE 1.2.6 REFERENCE ASME CODE PARAGRAPHS FOR HI-STORM FW OVERPACK and HI-TRAC VW TRANSFER CASK, PRIMARY LOAD BEARING PARTS			
	Item	Code Paragraph <sup>†</sup>	Notes, Explanation and Applicability
1.	Definition of primary and secondary members	NF-1215	-
2.	Jurisdictional boundary	NF-1133	The “intervening elements” are termed interfacing SSCs in this FSAR.
3.	Certification of material	NF-2130 (b) and (c)	Materials for ITS components shall be certified to the applicable Section II of the ASME Code or equivalent ASTM Specification.
4.	Heat treatment of material	NF-2170 and NF-2180	-
5.	Storage of welding material	NF-2440, NF-4411	-
6.	Welding procedure specification	Section IX	Acceptance Criteria per Subsection NF
7.	Welding material	Section II	-
8.	Definition of Loading conditions	NF-3111	-
9.	Allowable stress values	NF-3112.3	-
10.	Rolling and sliding supports	NF-3124	-
11.	Differential thermal expansion	NF-3127	-
12.	Stress analysis	NF-3143 NF-3380 NF-3522 NF-3523	Provisions for stress analysis for Class 3 linear structures is applicable for overpack top lid and the overpack and transfer cask shells.
13.	Cutting of plate stock	NF-4211 NF-4211.1	-
14.	Forming	NF-4212	-
15.	Forming tolerance	NF-4221	All cylindrical parts.
16.	Fitting and Aligning Tack Welds	NF-4231 NF-4231.1	-
17.	Alignment	NF-4232	-
18.	Cleanliness of Weld Surfaces	NF-4412	Applies to structural and non-structural welds
19.	Backing Strips, Peening	NF-4421 NF-4422	Applies to structural and non-structural welds
20.	Pre-heating and Interpass Temperature	NF-4611 NF-4612 NF-4613	Applies to structural and non-structural welds
21.	Non-Destructive Examination	NF-5360	Invokes Section V, Applies to Code welds only
22.	NDE Personnel Certification	NF-5522 NF-5523 NF-5530	Applies to Code welds only

<sup>†</sup> All references to the ASME Code refer to applicable sections of the 2007 edition.



TABLE 1.2.7  
SUMMARY REQUIREMENTS FOR MANUFACTURING  
OF HI-STORM FW SYSTEM COMPONENTS

	Item	MPC	HI-STORM FW	HI-TRAC VW Transfer Cask
1.	Material Specification	NB-2000 and ASME Section II	ASME Section II	ASME Section II
2.	Pre-welding operations (viz., cutting, forming, and machining)	NB-4000	Holtec Standard Procedures (HSPs)	Holtec Standard Procedures (HSPs)
3.	Weld wire	NB-2000 and ASME Section II	ASME Section II	ASME Section II
4.	Welding Procedure specifications and reference code for acceptance criteria	ASME Section IX and NB-4000	ASME Section IX and ASME Section III, Subsection NF	ASME Section IX
5.	NDE Procedures and reference code for acceptance criteria	ASME Section V, Subsection NB	ASME Section V, Subsection NF	ASME Section V, Subsection NF
6.	Qualification Protocol for Inspection Personnel	SNT-TC-1A	SNT-TC-1A	SNT-TC-1A
7.	Cleaning	ANSI N45.2.1 Section 2	ANSI N45.2.1 Section 2	ANSI N45.2.1 Section 2
8.	Packaging & Shipping	ANSI N45.2.2	ANSI N45.2.2	ANSI N45.2.2
9.	Mix or Plain Concrete	N/A	ACI 318 (2005)	N/A
10.	Inspection and Acceptance	Section 1.5 Drawings and Chapter 10	Section 1.5 Drawings and Chapter 10	Section 1.5 Drawings and Chapter 10
11.	Quality Procedures	Holtec Quality Assurance Procedures Manual	Holtec Quality Assurance Procedures Manual	Holtec Quality Assurance Procedures Manual

TABLE 1.2.8a [Withheld in Accordance with 10 CFR 2.390]
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TABLE 1.2.8b  
**[Withheld in Accordance with 10 CFR 2.390]**

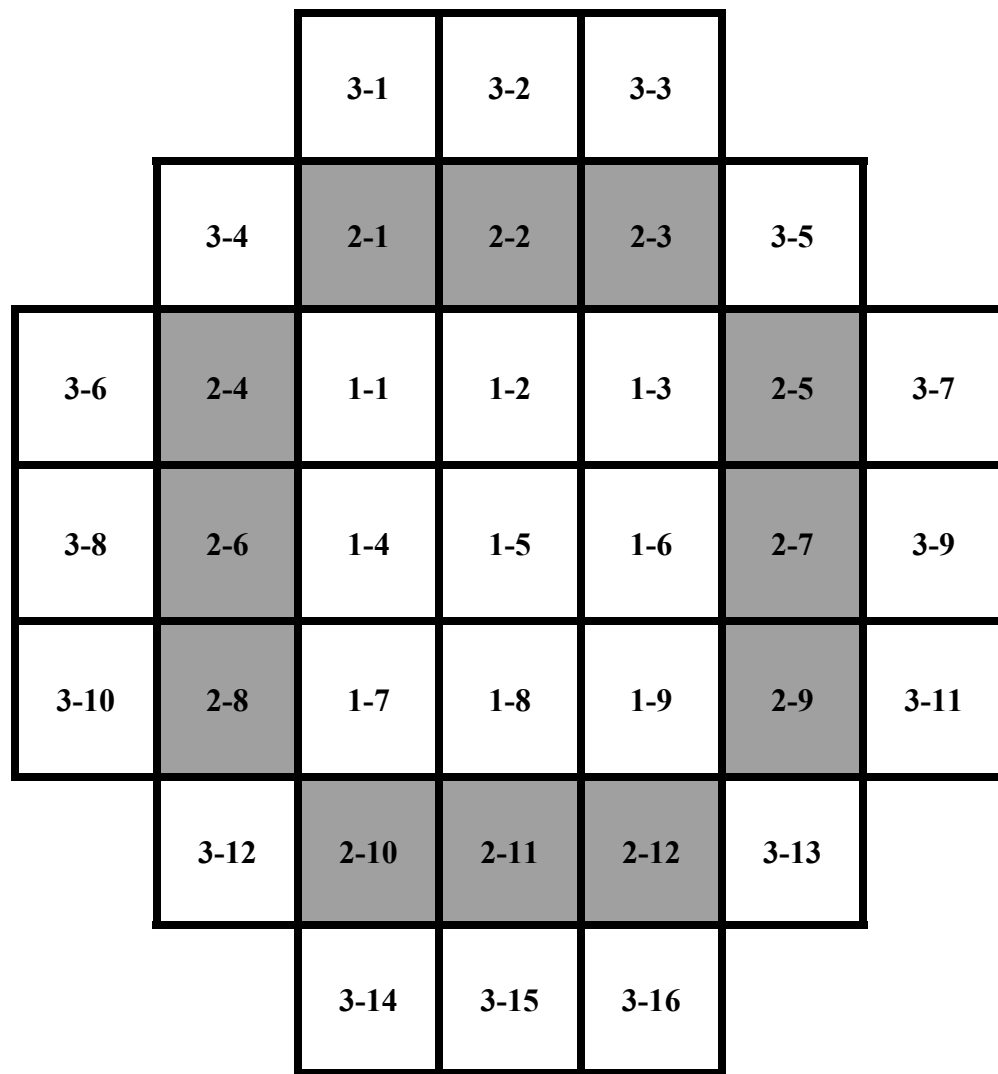
TABLE 1.2.9

## DESIGN OPTIONS FOR EXTRUDED BASKET SHIMS

Option	Emissivity of Extruded Basket Shims	As-Built Average Cold Radial Gap <sup>2</sup> (inch)	Solid Shims <sup>3</sup>	Emissivity of Solid Shims	Maximum Average Cold Radial Gap After Solid Shims are Placed (inch) <sup>4</sup>
1	Note 1	$\leq 0.281$	NOT REQUIRED	Not Applicable	Not Applicable
2	Note 1	$> 0.281$	REQUIRED	Note 1	$\leq 0.24$
3	0.4	$\leq 0.2$	NOT REQUIRED	Not Applicable	Not Applicable
4	0.4	$> 0.2$	REQUIRED	Note 1	$\leq 0.2$
5	0.6	$\leq 0.281$	NOT REQUIRED	Note Applicable	Not Applicable
6	0.6	$>0.281$	REQUIRED	Note 1	$\leq 0.24$

## Notes:

1. Emissivity must meet the requirements tabulated in Note 2 of Table 1.2.8.
2. This is the average total combined radial cold air gap between the basket and extruded shims, and the extruded shim and the inner surface of the MPC shell before the placement of solid shim plates.
3. Extruded shims are shaped to conform to the geometry of its intended annular space and sized to provide a loose fit in the basket periphery. If the as-built average total combined radial cold gap between the basket and extruded shims and the extruded shim and the inner surface of the MPC enclosure shell exceeds the gap tabulated herein, solid shim aluminum plates shall be inserted in the space between the basket external wall and extruded shims.
4. The average total combined radial cold air gap between the basket and extruded shims and the extruded shim and the inner surface of the MPC shell must be below the value tabulated herein if solid shim plates are placed between the basket wall and extruded shim.

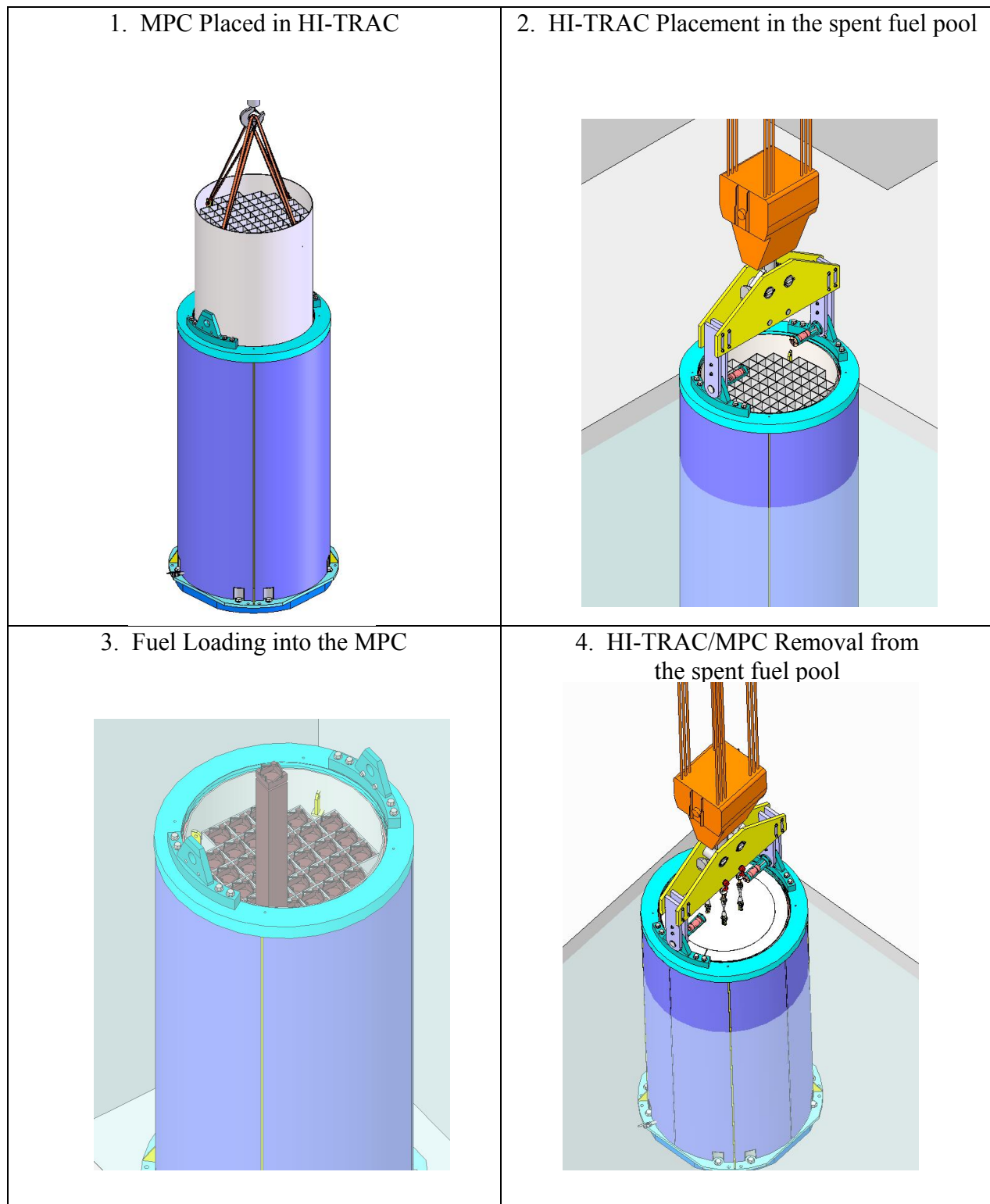
**Legend**

Region- Cell ID
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Figure 1.2.1: MPC-37 Basket, Region and Cell Identification

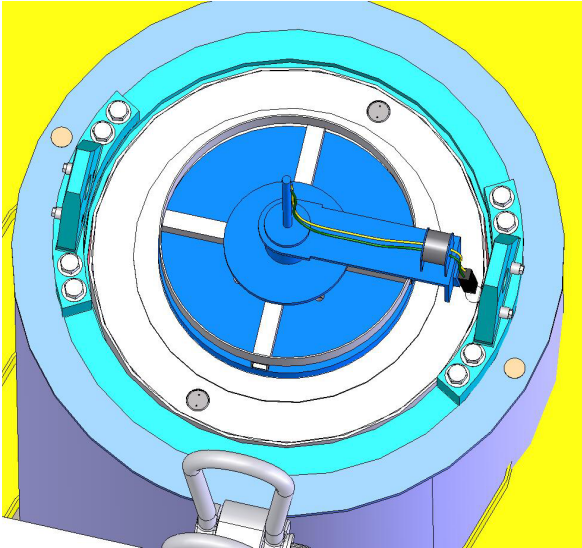
**Legend**

**Region-  
Cell ID**

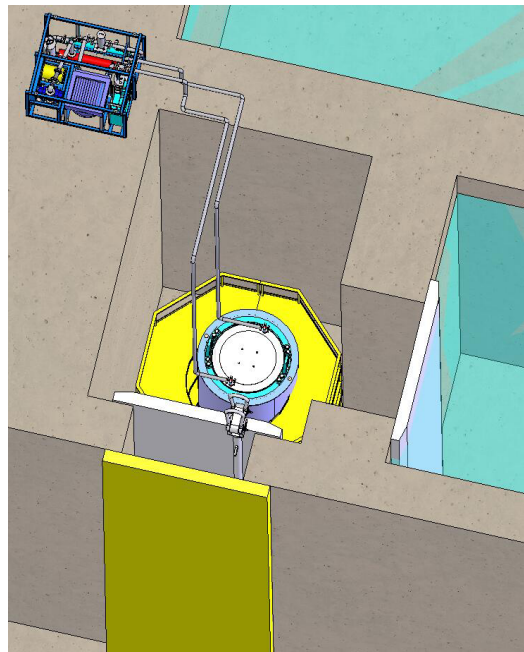


**FIGURE 1.2.3: SUMMARY OF TYPICAL LOADING OPERATIONS**

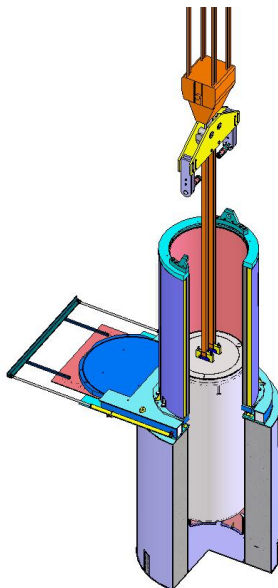
5. MPC Closure Operations  
(Lid to Shell Welding)



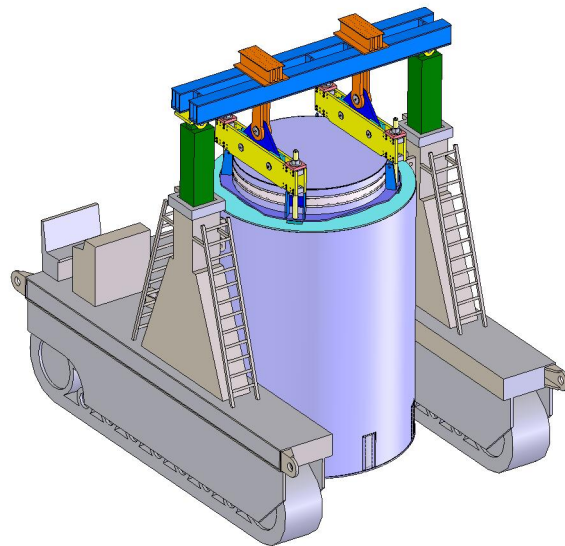
6. MPC Draining, Drying and Backfill



7. System Stackup and MPC Transfer Operations



8. HI-STORM Movement to the ISFSI



**FIGURE 1.2.3 (CONTINUED): SUMMARY OF TYPICAL  
LOADING OPERATIONS**



### 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, headquartered in Marlton, NJ, is the system designer and applicant for certification of the HI-STORM FW 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 80 plants in the U.S., Britain, Brazil, Korea, Mexico and Taiwan have utilized the Company's wet storage technology to extend their in-pool storage capacities.

NPD is also a turnkey provider of dry storage and transportation technologies to nuclear plants around the globe. The company is contracted by over 40 nuclear units in the U.S. to provide the company's vertical ventilated dry storage technology. Utilities in China, Korea, Spain, Ukraine, 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-STORM FW) 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 2009) are listed in Table 1.3.2. Many of these listed patents have been utilized in the design of the HI-STORM FW System.

Holtec International's quality assurance (QA) program was originally developed to meet NRC requirements delineated in 10CFR50, Appendix B, and was expanded to include provisions of 10CFR71, 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

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incorporated by reference into this FSAR. Holtec International's QA program has been certified by the USNRC (Certificate No. 71-0784).

The HI-STORM FW System will be fabricated by Holtec International Manufacturing Division (HMD) located in Pittsburgh, Pennsylvania. HMD is a long term N-Stamp holder and fabricator of nuclear components. In particular, HMD has been manufacturing HI-STORM and HI-STAR system components since the inception of Holtec International's dry storage and transportation program in the 1990s. HMD routinely manufactures ASME code components for use in the US and overseas nuclear plants. Both Holtec International's headquarters and the HMD subsidiary have been subject to triennial inspections by the USNRC. If another fabricator is to be used for the fabrication of any part of the HI-STORM FW System, the proposed fabricator will be evaluated and audited in accordance with Holtec International's QA program.

The Metamic-HT will be fabricated by Holtec International Nanotec Division (Nanotec) located in Lakeland, Florida. Nanotec has been manufacturing classic Metamic for several years for both dry and wet storage applications and in the last few years has been manufacturing and testing Metamic-HT. If another fabricator is to be used for the fabrication of Metamic-HT, the proposed fabricator will be evaluated and audited in accordance with Holtec International's QA program.

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 (completed) and Waterford (ongoing, ca. 2009) have deployed dry storage at their sites using a turn key contract with Holtec International.

TABLE 1.3.1	
USNRC DOCKETS ASSIGNED TO HOLTEC INTERNATIONAL	
<b>System Name</b>	<b>Docket Number</b>
HI-STORM 100 (Storage)	72-1014
HI-STAR 100 (Storage)	72-1008
HI-STORM Flood/Wind (Storage)	72-1032
HI-STORM UMAX (Storage)	72-1040
HI-STAR 100 (Transportation)	71-9261
HI-STAR 180 (Transportation)	71-9325
HI-STAR 180D (Transportation)	71-9367
HI-STAR 60 (Transportation)	71-9336
Holtec Quality Assurance Program	71-0784

TABLE 1.3.2	
DRY STORAGE AND TRANSPORT PATENTS ASSIGNED TO HOLTEC INTERNATIONAL	
Colloquial Name of the patent	USPTO Patent Number
Honeycomb Fuel Basket	5,898,747
HI-STORM 100S Overpack	6,064,710
Duct Photon Attenuator	6,519,307B1
HI-TRAC Operation	6,587,536B1
Cask Mating Device (Hermetically Sealable Transfer Cask)	6,625,246B1
Improved Ventilator Overpack	6,718,000B2
Below Grade Transfer Facility	6,793,450B2
HERMIT (Seismic Cask Stabilization Device)	6,848,223B2
Cask Mating Device ( operation)	6,853,697
Davit Crane	6,957,942B2
Duct-Fed Underground HI-STORM	7,068,748B2
Forced Helium Dehydrator (design)	7,096,600B2
Below Grade Cask Transfer Facility	7,139,358B2
Forced Gas Flow Canister Dehydration (alternate embodiment)	7,210,247B2
HI-TRAC Operation (Maximizing Radiation Shielding During Cask Transfer Procedures)	7,330,525
HI-STORM 100U	7,330,526B2

## 1.4 GENERIC CASK ARRAYS

The HI-STORM FW System is stored in a vertical configuration. The required center-to-center spacing between the modules (layout pitch) on the Independent Spent Fuel Storage Installation (ISFSI) pad is guided by operational considerations such as size, accessibility, security, dose, and functionality. Tables 1.4.1, 1.4.2, and 1.4.3 provide the typical layout pitch information for 2 x N (N can be any integer), 3 x N (N can be any integer), and rectangular arrays, respectively.

The following is a generic discussion on the HI-STORM FW ISFSI pad, its suggested arrangement, and supporting infrastructure. The final design of the ISFSI is the responsibility of the user of the HI-STORM FW System.

The HI-STORM FW ISFSI pad is typically 24" to 28" thick, reinforced concrete supported by engineered fill with depth and properties selected to satisfy a site-specific design. The casks are arrayed in the manner of a rectilinear grid such as that shown in Figures 1.4.1, 1.4.2, and 1.4.3. The pitch values in Table 1.4.1 may be varied to suit the user's specific needs. The spacing (X, Y, etc., in the figures) is chosen to satisfy two competing requirements. Typically, the ISFSI owner desires to minimize the spacing in order to produce self-shielding between the storage casks, however the spacing must also be sufficient to allow the transporter access to emplace and remove the overpacks. The HI-STORM FW spacing (pitch) shown in Table 1.4.1 are typical values that meet both competing requirements.

A Canister Transfer Facility (CTF) may be needed in the future (when the Fuel Building is no longer available) to remove the multi-purpose canister from the HI-STORM FW overpack and place it into a HI-STAR transport cask, suitable for offsite shipment. The MPC transfer should be performed in a controlled area. Therefore, the ISFSI facility should preferably be sized to accommodate the CTF; however the construction of the CTF can be performed during a later development phase.

The general area surrounding the HI-STORM FW ISFSI pad will be graded to be compatible with the current drainage features, with additional storm water catch basins and piping added and incorporated into the existing storm water collection system, as necessary. The general area surrounding the ISFSI pad is typically covered with crushed stone or gravel to provide a suitable surface for the transporter and to prevent weeds and other unsuitable foliage from sprouting.

The ISFSI should have an area designated as a HI-STORM FW fabrication pad. This area is used to prepare HI-STORM FW casks for concrete placement, assembly, touch-up painting, storage, and maintenance between the time of initial on-site delivery and actual MPC transfer. An adjacent garage and maintenance shop may also be required for housing the transfer cask and ancillaries, such as the transporter, lifting appurtenances, etc.

If the ISFSI pad is located outside the plant's protected area, a security post building to provide a weather enclosure for temporary security guard support staff may be needed during casks movement and facility access. The building would also provide a common termination point for security equipment wiring and the HI-STORM FW temperature monitoring data acquisition equipment, if used. A backup power diesel generator and associated transformers may be skid mounted on a pad

adjacent to the security post.

The discussion of the security and related systems below presumes that the ISFSI is located outside the plant's protected area. The security requirements are adjusted accordingly if the ISFSI is located inside the plant's protected area.

The requirements on the security system provided below are generic and illustrative of the state-of-the-art practice, i.e., they are not meant to be mandatory provisions. The ISFSI owner bears the ultimate responsibility to comply with all security related regulations and mandates.

#### 1.4.2 Security System and Other Ancillary Requirements

A security system for the ISFSI will be designed to include intrusion detection and camera systems, security fencing, lighting, isolation zones, monitoring systems, and electrical supply. The design must be integrated with the existing plant security system and its components. The system must meet the requirements of 10CFR72 and 10CFR73, and shall be integrated into the existing Plant's Physical Security Plan. The design of the security system shall also take into consideration the guidelines provided by NUREG-1619, NUREG-1497, and NRC Regulatory Guide 5.44.

Electrical design features must also be included for HI-STORM FW temperature monitoring, HI-STORM FW grounding, and the storage/maintenance building, as required. The HI-STORM FW temperature monitoring system (if used) will include thermal detectors mounted directly to the overpacks. These detectors will provide continuous monitoring and data acquisition equipment to collect, process, and transmit data to a central computer system to allow frequent review of data results and to indicate any temperature alerts. The storage building should have sufficient electrical power supply to support lights, outlets, and power equipment associated with maintenance of HI-STORM FW ancillary equipment, such as the transporter. In the event of loss of power to the site, a backup power supply is required.

##### 1.4.2.1 Security System

The ISFSI security system design shall provide the layout for all components and associated power and signal wiring. The security interface building located adjacent to the ISFSI would provide a transition point to connect all of the wiring to the existing plant power and data acquisition systems.

The ISFSI security systems will consist of two separate systems supplementing each other: perimeter intrusion detection system (PIDS) and a closed circuit television (CCTV) system. The PIDS will provide an alarm signal to the existing security system whenever one of the perimeter zones has been accessed without authorization. The CCTV system will provide assessment of the alarming zone. Both of these systems have to work with each other in order to provide proper assessment. All signals generated by the security systems will be transmitted to the Central Alarm Station (CAS) through a robust communication means. The ISFSI security system design will be compatible with the plant's existing design.

The security systems design will include details for PIDS mounting, CCTV system mounting, zone arrangements, fiber optic hardware/cable connections for alarm and tamper, camera and microwave

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unit locations, and upgrades to the existing security system to accommodate the new ISFSI systems.

#### **1.4.2.2 Lighting System**

The design of the lighting system includes light fixture selection, quantity, mounting, and arrangement throughout ISFSI perimeter and the assessment of illumination levels in foot-candles.

The illumination levels required at the perimeter area and inside the plant's protected area will be maintained at the ISFSI in accordance with plant commitments and regulatory requirements. The design will also include infrared illuminators to be installed, as an option with the CCTV system cameras to provide minimum light level required for IR sensitive cameras.

#### **1.4.2.3 Fence System**

The design for ISFSI perimeter fence includes a double fence configuration. The inner fence will be the protected area perimeter and the outer fence will be a nuisance fence to establish the appropriate isolation zone. The typical fence arrangements, including man-gates; vehicle gates; and grounding details; will be based on the existing plant fence specifications and design standards.

#### **1.4.2.4 Electrical System**

The conceptual design for the electrical system would entail the following activities and use their results as inputs:

- design for security systems (PIDS and CCTV)
- design for perimeter lighting system (PLS)
- design for temperature monitoring system (TMS) (if used)
- design for storage/support building

The total ISFSI site load will determine what type and size of power source will be used in this application. The existing power distribution facilities must be reviewed to determine a capability of the potential power sources. To be able to add the new ISFSI load to an existing system an analysis will be completed including the evaluation of the existing loads on 4160VAC line, cable sizes, and the approximate cable length. The transformers (4160-480V and 480-208/120V) will be sized accordingly to accommodate a new distribution system. The conceptual design will also include all the aspects of sizing a backup power distribution system based on providing a dedicated diesel generator as a source.

#### **1.4.2.5 Cask Grounding System**

The design of the grounding system should be based on NEC requirements and engineering and plant practices. The new grounding system, if required, will surround the ISFSI perimeter and provide a ground path for all ISFSI related equipment and structures including storage casks, microwave equipment and mounting poles, camera and towers, security lighting, perimeter fences, and the security building at the ISFSI site. The grounding system will be connected to the primary source transformer ground.

TABLE 1.4.1

TYPICAL (AND MINIMUM) LAYOUT PITCH AND SPACING DIMENSIONS FOR HI-STORM FW ARRAYS

<b>Item</b>	<b>Layout in Figure 1.4.1</b>	<b>Layout in Figure 1.4.2</b>	<b>Layout in Figure 1.4.3</b>
X1	16 ft (15 ft)	16 ft (15 ft)	16 ft (15 ft)
Y1	16 ft (15 ft)	16 ft (15 ft)	16 ft (15 ft)
Y2	12 ft	12 ft	N/A
Y3	12 ft	12 ft	N/A



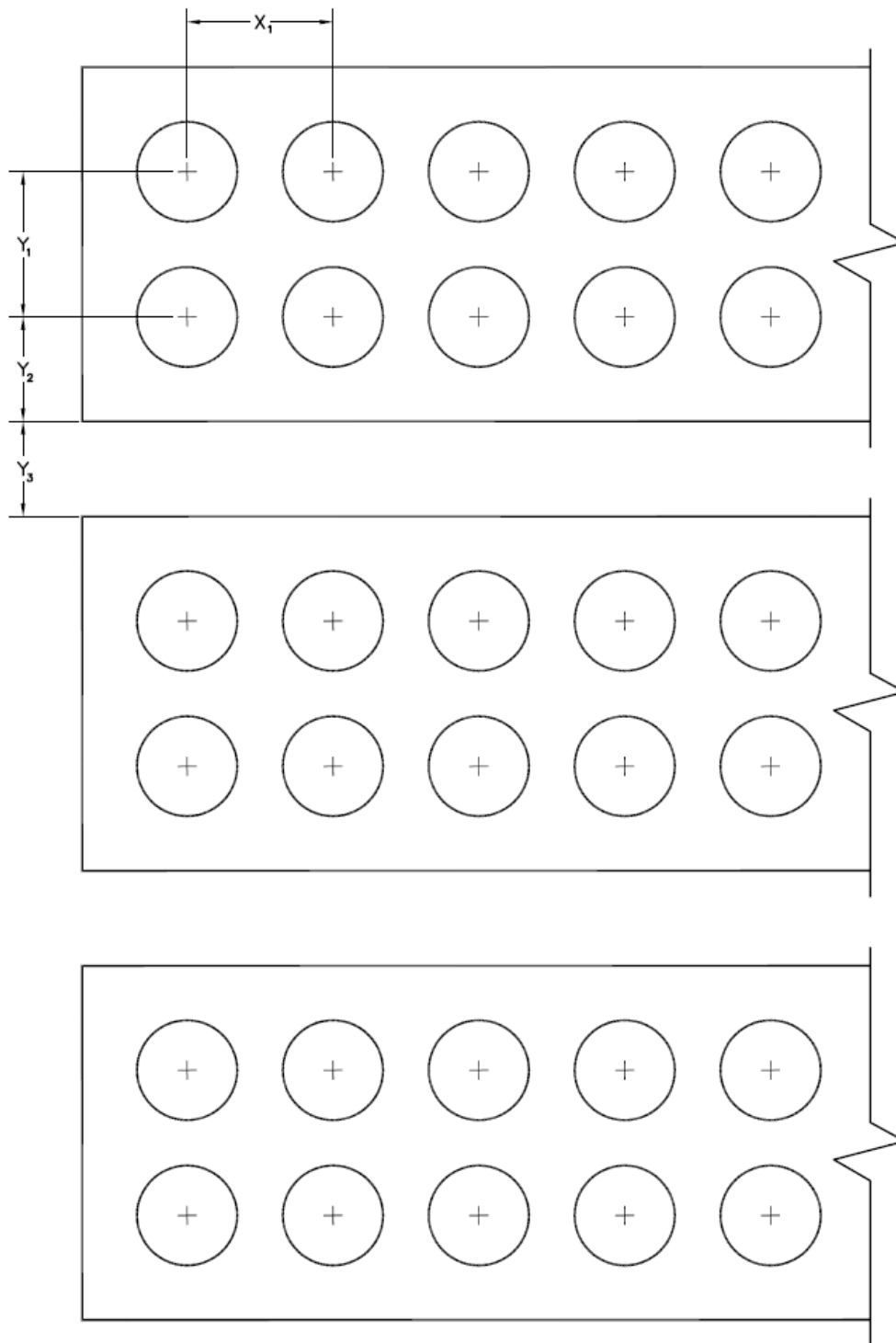


FIGURE 1.4.1: 2xN HI-STORM FW ARRAYS

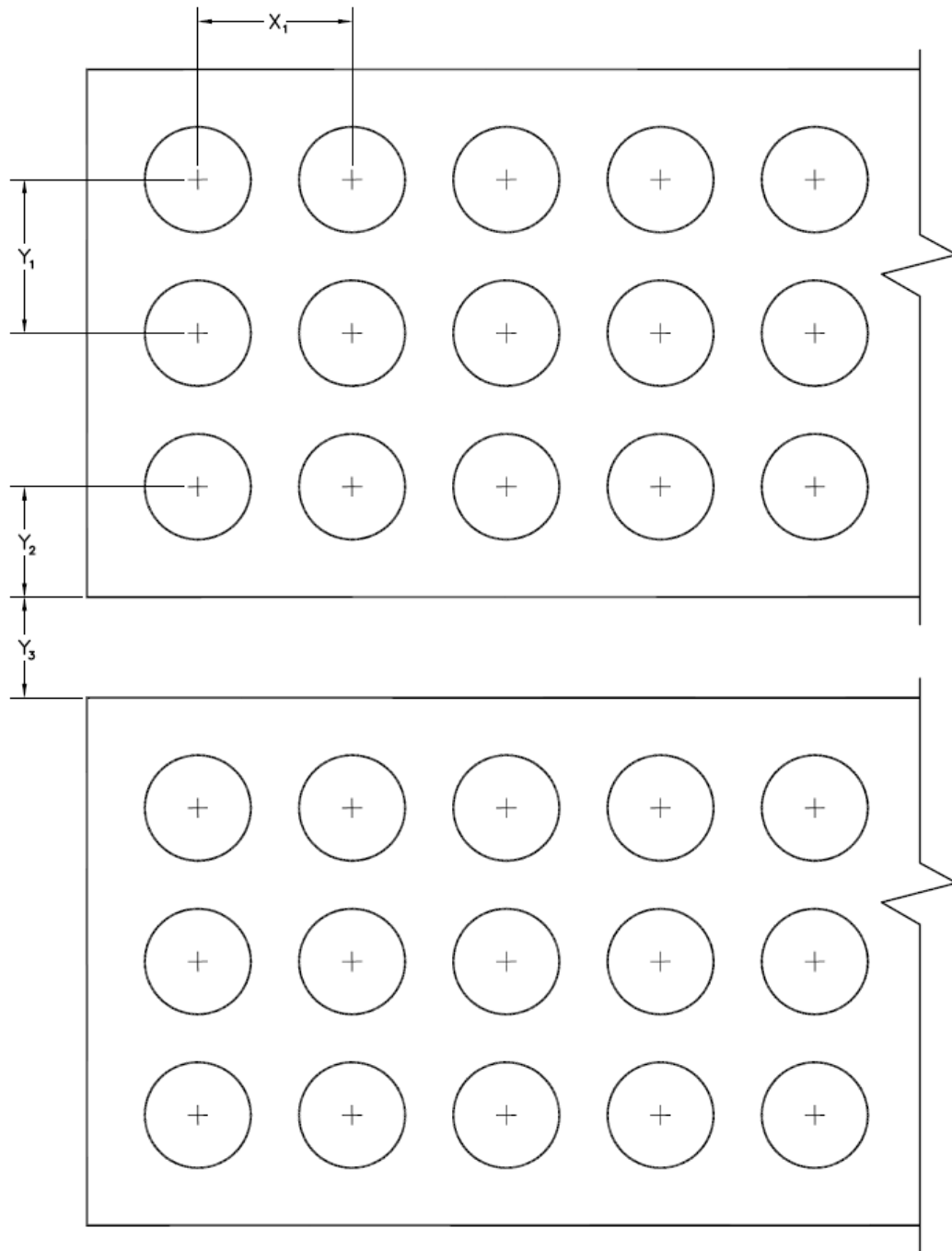


FIGURE 1.4.2: 3xN HI-STORM FW ARRAYS

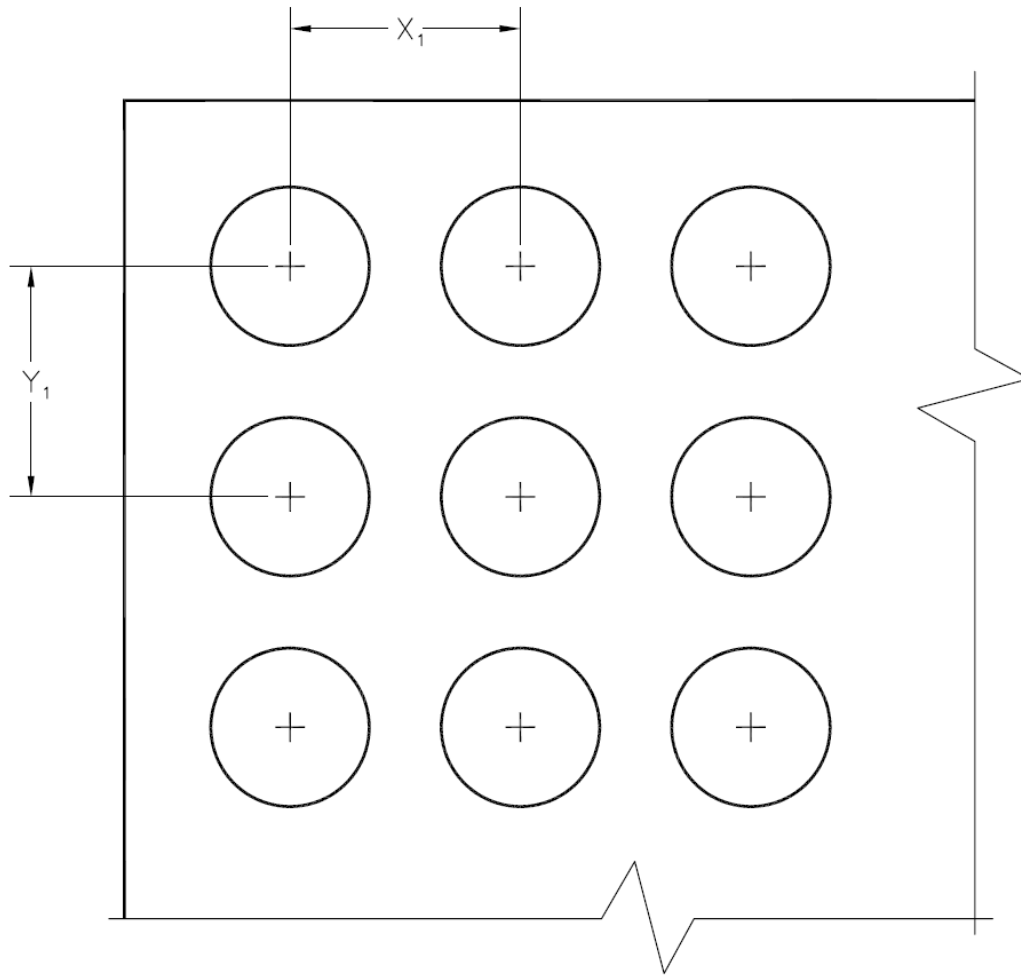


FIGURE 1.4.3: RECTANGULAR HI-STORM FW ARRAY

## 1.5 DRAWINGS

The following HI-STORM FW System drawings are provided on subsequent pages in this section to fulfill the requirements in 10 CFR 72.2(a)(1),(b) and 72.230(a):

Drawing No.	Title	Revision
6494	HI-STORM FW BODY	12
6508	HI-STORM LID ASSEMBLY	8
6514	HI-TRAC VW – MPC-37	9
6799	HI-TRAC VW – MPC-89	9
6505	MPC-37 ENCLOSURE VESSEL	13
6506	MPC-37 FUEL BASKET	12
6512	MPC-89 ENCLOSURE VESSEL	15
6507	MPC-89 FUEL BASKET	11

**[Withheld in Accordance with 10 CFR 2.390]**

## 1.6 REFERENCES

- [1.0.1] 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Fuel, High-level Radioactive Waste, and Reactor-Related Greater than Class C Waste”, Title 10 of the Code of Federal Regulations- Energy, Office of the Federal Register, Washington, D.C.
- [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] Regulatory Guide 1.76 “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plant”, U.S. Nuclear Regulatory Commission, March 2007.
- [1.1.1] ASME Boiler & Pressure Vessel Code, Section III, Subsection NB, American Society of Mechanical Engineers, New York, 2007.
- [1.1.2] 10CFR Part 50, “Domestic Licensing of Production and Utilization Facilities”, Title 10 of the Code of Federal Regulations, Office of the Federal Register, Washington, D.C.
- [1.1.3] USNRC Docket 72-1014, “Final Safety Analysis Report for the HI-STORM 100 System”, Holtec Report No. HI-2002444, latest revision.
- [1.1.4] NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety”, U.S. Nuclear Regulatory Commission, February 1996.
- [1.2.1] U.S. NRC Information Notice 96-34, “Hydrogen Gas Ignition During Closure Welding of a VSC-24 Multi-Assembly Sealed Basket”.
- [1.2.2] American Concrete Institute, “Building Code Requirements for Structural Plain Concrete (ACI 318.1-89) (Revised 1992) and Commentary - ACI 318.1R-89 (Revised 1992)”.
- [1.2.3] ANSI N14.6-1993, “American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 Kg) or More”, American National Standards Institute, Inc., Washington D.C., June 1993.
- [1.2.4] Companion Guide to the ASME Boiler & Pressure Vessel Code, K.R. Rao (editor), Chapter 56, “ Management of Spent Nuclear Fuel”, Third Edition, ASME (2009).
- [1.2.5] HI-STAR 180 Transportation Package, USNRC Docket No. 71-9325.

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- [1.2.6] “Metamic-HT Qualification Sourcebook”, Holtec Report No. HI-2084122, Latest Revision (Holtec Proprietary).
- [1.2.7] “Metamic-HT Manufacturing Manual”, Nanotec Metals Division, Holtec International, Latest Revision (Holtec Proprietary).
- [1.2.8] “Metamic-HT Purchasing Specification”, Holtec Document ID PS-11, Latest Revision, (Holtec Proprietary).
- [1.2.9] Sampling Procedures and Tables for Inspection by Attributes”, Military Standard MIL-STD-105E, (10/5/1989).
- [1.2.10] USNRC Docket No. 72-1004 SER on NUHOMS 61BT (2002).
- [1.2.11] Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Holtec International Report HI-2022871 Regarding Use of Metamic in Fuel Pool Applications,” Facility Operating License Nos. DPR-51 and NPF-6, Entergy Operations, Inc., docket No. 50-313 and 50-368, USNRC, June 2003.
- [1.2.12] Dynamic Mechanical Response and Microstructural Evolution of High Strength Aluminum-Scandium (Al-Sc) Alloy, by W.S. Lee and T.H. Chen, Materials Transactions, Vol. 47, No. 2(2006), pp 355-363, Japan Institute for metals.
- [1.2.13] Turner, S.E., “Reactivity Effects of Streaming Between Discrete Boron Carbide Particles in Neutron Absorber Panels for Storage or Transport of Spent Nuclear Fuel,” Nuclear Science and Engineering, Vol. 151, Nov. 2005, pp. 344-347.
- [1.2.14] Natrella, M.G., “Experimental Statistics”, National Bureau of Standards Handbook 91, National Bureau of Standards, Washington, DC, 1963.
- [1.2.15] Technical Memo TM-141R1, NRC’s Guidance on Design of Lifting Systems and Special Lifting Devices Used in Holtec’s Used Fuel Management Program, dated 4/15/15.

## APPENDIX 1.A: ALLOY X DESCRIPTION

### 1.A.1 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 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 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 ( $\alpha$ )
- 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 -20°F. The lower bound service temperature of the MPC is -40°F. Most of the above-mentioned properties become increasingly favorable as the temperature drops. Conservatively, the values at the lowest design temperature for the HI-STORM FW 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°F is linearly extrapolated from the 70°F value using the difference from 70°F to 100°F.

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. 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, Materials (2007).



TABLE 1.A.1					
DESIGN STRESS INTENSITY ( $S_m$ ) vs. TEMPERATURE FOR THE ALLOY-X MATERIALS					
Temp. (°F)	Type 304	Type 304LN	Type 316	Type 316LN	Alloy X (minimum of constituent values)
-40	20.0	20.0	20.0	20.0	20.0
100	20.0	20.0	20.0	20.0	20.0
200	20.0	20.0	20.0	20.0	20.0
300	20.0	20.0	20.0	20.0	20.0
400	18.6	18.6	19.3	18.9	18.6
500	17.5	17.5	18.0	17.5	17.5
600	16.6	16.6	17.0	16.5	16.5
650	16.2	16.2	16.6	16.0	16.0
700	15.8	15.8	16.3	15.6	15.6
750	15.5	15.5	16.1	15.2	15.2
800	15.2	15.2	15.9	14.8	14.8

Notes:

1. Source: Table 2A on pages 308, 312, 316, and 320 of [1.A.1].
2. Units of design stress intensity values are ksi.

TABLE 1.A.2					
TENSILE STRENGTH ( $S_u$ ) vs. TEMPERATURE OF ALLOY-X MATERIALS					
Temp. (°F)	Type 304	Type 304LN	Type 316	Type 316LN	Alloy X (minimum of constituent values)
-40	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)
100	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)	75.0 (70.0)
200	71.0 (66.3)	71.0 (66.3)	75.0 (70.0)	75.0 (70.0)	71.0 (66.3)
300	66.2 (61.8)	66.2 (61.8)	72.9 (68.0)	70.7 (66.0)	66.2 (61.8)
400	64.0 (59.7)	64.0 (59.7)	71.9 (67.1)	67.1 (62.6)	64.0 (59.7)
500	63.4 (59.2)	63.4 (59.2)	71.8 (67.0)	64.6 (60.3)	63.4 (59.2)
600	63.4 (59.2)	63.4 (59.2)	71.8 (67.0)	63.3 (59.0)	63.3 (59.0)
650	63.4 (59.2)	63.4 (59.2)	71.8 (67.0)	62.8 (58.6)	62.8 (58.6)
700	63.4 (59.2)	63.4 (59.2)	71.8 (67.0)	62.4 (58.3)	62.4 (58.3)
750	63.3 (59.0)	63.3 (59.0)	71.5 (66.7)	62.1 (57.9)	62.1 (57.9)
800	62.8 (58.6)	62.8 (58.6)	70.8 (66.1)	61.7 (57.6)	61.7 (57.6)

Notes:

1. Source: Table U on pages 514, 516, 518, 520, and 522 of [1.A.1].
2. Units of tensile strength are ksi.
3. The ultimate stress of Alloy X is dependent on the product form of the material (i.e., forging vs. plate). Values in parentheses 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.

TABLE 1.A.3					
YIELD STRESSES ( $S_y$ ) vs. TEMPERATURE OF ALLOY-X MATERIALS					
Temp. ( $^{\circ}$ F)	Type 304	Type 304LN	Type 316	Type 316LN	Alloy X (minimum of constituent values)
-40	30.0	30.0	30.0	30.0	30.0
100	30.0	30.0	30.0	30.0	30.0
200	25.0	25.0	25.9	25.5	25.0
300	22.4	22.4	23.4	22.9	22.4
400	20.7	20.7	21.4	21.0	20.7
500	19.4	19.4	20.0	19.5	19.4
600	18.4	18.4	18.9	18.3	18.3
650	18.0	18.0	18.5	17.8	17.8
700	17.6	17.6	18.2	17.3	17.3
750	17.2	17.2	17.9	16.9	16.9
800	16.9	16.9	17.7	16.5	16.5

Notes:

1. Source: Table Y-1 on pages 634, 638, 646, and 650 of [1.A.1].
2. Units of yield stress are ksi.

TABLE 1.A.4	
COEFFICIENT OF THERMAL EXPANSION vs. TEMPERATURE OF ALLOY-X MATERIALS	
Temp. (°F)	Type 304, 304LN, 316, 316LN
-40	--
100	8.6
150	8.8
200	8.9
250	9.1
300	9.2
350	9.4
400	9.5
450	9.6
500	9.7
550	9.8
600	9.8
650	9.9
700	10.0
750	10.0
800	10.1
850	10.2
900	10.2
950	10.3
1000	10.3
1050	10.4
1100	10.4

Notes:

1. Source: Group 3 alloys from Table TE-1 on pages 749 and 751 of [1.A.1].
2. Units of mean coefficient of thermal expansion are in./in./°F x 10<sup>-6</sup>.

TABLE 1.A.5 THERMAL CONDUCTIVITY vs. TEMPERATURE OF ALLOY-X MATERIALS			
Temp. (°F)	Type 304 and Type 304LN	Type 316 and Type 316LN	Alloy X (minimum of constituent values)
-40	--	--	--
70	8.6	8.2	8.2
100	8.7	8.3	8.3
150	9.0	8.6	8.6
200	9.3	8.8	8.8
250	9.6	9.1	9.1
300	9.8	9.3	9.3
350	10.1	9.5	9.5
400	10.4	9.8	9.8
450	10.6	10.0	10.0
500	10.9	10.2	10.2
550	11.1	10.5	10.5
600	11.3	10.7	10.7
650	11.6	10.9	10.9
700	11.8	11.2	11.2
750	12.0	11.4	11.4
800	12.3	11.6	11.6
850	12.5	11.9	11.9
900	12.7	12.1	12.1
950	12.9	12.3	12.3
1000	13.1	12.5	12.5
1050	13.4	12.8	12.8
1100	13.6	13.0	13.0

## Notes:

1. Source: Material groups J and K in Table TCD on page 765, 766, and 775 of [1.A.1].
2. Units of thermal conductivity are Btu/hr-ft-°F.

## CHAPTER 2<sup>†</sup>: PRINCIPAL DESIGN CRITERIA

### 2.0 INTRODUCTION

The design characteristics of the HI-STORM FW System are presented in Chapter 1, Section 1.2. This chapter contains a compilation of loadings and design criteria applicable to the HI-STORM FW System. The loadings and conditions prescribed herein for the MPC, particularly those pertaining to mechanical accidents, are consistent with those required for 10CFR72 compliance. 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 structural design criteria are presented in the subsequent chapters of this FSAR.

This chapter is in full compliance with NUREG-1536, with the exceptions and clarifications provided in Table 1.0.3. Table 1.0.3 summarizes the NUREG-1536 review guidance, the justification for the exception or clarification, and the Holtec approach to meet the intent of the NUREG-1536 guidance.

The design criteria for the MPCs, HI-STORM FW overpack, and HI-TRAC VW transfer cask are summarized in Subsections 2.0.1, 2.0.2, and 2.0.3, respectively, and described in the sections that follow.

#### 2.0.1 MPC Design Criteria

##### General

The MPC is engineered for a 60 year design life, while satisfying the requirements of 10CFR72. The adequacy of the MPC to meet the above design life is discussed in Section 3.4. The design characteristics of the MPC are described in Section 1.2.

##### Structural

The MPC is classified as important-to-safety. The MPC structural components include the fuel basket and the enclosure vessel. The fuel basket is designed and fabricated to meet a more stringent displacement limit under mechanical loadings than those implicit in the stress limits of the ASME code (see Section 2.2). The MPC enclosure vessel is designed and fabricated as a Class 1 pressure vessel in accordance with Section III, Subsection NB of the ASME Code, with

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<sup>†</sup> This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. The material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. All terms-of-art used in this chapter are consistent with the terminology of the Glossary.

certain necessary alternatives, as discussed in Section 2.2. The principal exception to the above Code pertains to the MPC lid, vent and drain port cover plates, and closure ring welds to the MPC lid and shell, as discussed in Section 2.2. In addition, Threaded Anchor Locations (TALs) in the MPC lid are designed in accordance with the requirements of NUREG-0612 for critical lifts to facilitate handling of the loaded MPC.

The MPC closure welds are partial penetration welds that are structurally qualified by analysis in Chapter 3. The MPC lid and closure ring welds are inspected by performing a liquid penetrant examination in accordance with the drawings contained in Section 1.5. The integrity of the MPC lid-to-shell weld is further verified by performing a progressive liquid penetrant examination of the weld layers, and a Code pressure test.

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

Compliance with the ASME Code, with respect to the design and fabrication of the MPC, and the associated justification are discussed in Section 2.2. The MPC design is analyzed for all design basis normal, off-normal, and postulated accident conditions, as defined in Section 2.2. The required characteristics of the fuel assemblies to be stored in the MPC are limited in accordance with Section 2.1.

### Thermal

The thermal design and operation of the MPC in the HI-STORM FW System meets the intent of the review guidance contained in ISG-11, Revision 3 [2.0.1]. 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 400°C (752°F). For short-term operations, including canister drying, helium backfill, and on-site cask transport operations, the fuel cladding temperature must not exceed 400°C (752°F) for high burnup fuel (HBF) and 570°C (1058°F) for moderate burnup fuel.
- iii. The maximum fuel cladding temperature as a result of an off-normal or accident event must not exceed 570°C (1058°F).
- iv. For HBF, operating restrictions are imposed to limit the maximum temperature excursion during short-term operations to 65°C (117°F) 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 400°C (752°F). However, for MPCs containing all moderate burnup fuel, the fuel cladding temperature limit for short-term operations is set at 570°C (1058°F) because the nominal fuel cladding stress is shown to be less than 90 MPa [2.0.2]. 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 °C (1058°F).

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 normal condition design temperatures for the stainless steel components in the MPC are provided in Table 2.2.3.

Each MPC model allows for regionalized storage where the basket is segregated into three regions as shown in Figures 1.2.1 and 1.2.2. Decay heat limits for regionalized loading are presented in Tables 1.2.3 and 1.2.4 for MPC-37 and MPC-89, respectively. Specific requirements, such as approved locations for DFCs and non-fuel hardware are given in Section 2.1.

### Shielding

The dose limits for an ISFSI using the HI-STORM FW System are delineated in 10CFR72.104 and 72.106. Compliance with these regulations for any particular array of casks at an ISFSI is necessarily site-specific and must be demonstrated by the licensee. Dose for a single cask and a representative cask array is illustrated in Chapter 5.

The MPC provides axial shielding at the top and bottom ends to maintain occupational exposures ALARA during canister closure and handling operations. The HI-TRAC VW bottom lid also contains shielding. The occupational doses are controlled in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 9).

The dose evaluation is performed for a reference fuel (Table 1.0.4) as described in Section 5.2. Calculated dose rates for each MPC are provided in Section 5.1. These dose rates are used to perform an occupational exposure (ALARA) evaluation, as discussed in Chapter 11.



### Criticality

The MPC provides 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  $k_{\text{eff}} < 0.95$  for fresh (unirradiated) fuel with optimum water moderation and close reflection, including all biases, uncertainties, and manufacturing tolerances.

Criticality control is maintained by the geometric spacing of the fuel assemblies and the spatially distributed B-10 isotope in the Metamic-HT fuel basket, and for the PWR MPC model, the additional soluble boron in the MPC water. The minimum specified boron concentration in the purchasing specification for Metamic-HT must be met in every lot of the material manufactured. The guaranteed B-10 value in the neutron absorber, assured by the manufacturing process, is further reduced by 10% (90% credit is taken for the Metamic-HT) to accord with NUREG/CR-5661. No credit is taken for fuel burnup or integral poisons such as gadolinia in BWR fuel. The soluble boron concentration requirements (for PWR fuel only) based on the initial enrichment of the fuel assemblies are delineated in Section 2.1 consistent with the criticality analysis described in Chapter 6.

### 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-STORM FW MPC design meets the guidance in Interim Staff Guidance (ISG)-18 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 10.

### Operations

There are no radioactive effluents that result from storage or transfer operations. Effluents generated during MPC loading are handled by the plant's radioactive waste system and procedures.

Generic operating procedures for the HI-STORM FW System are provided in Chapter 9. Detailed operating procedures will be developed by the licensee using the information provided in Chapter 9 along with the site-specific requirements that comply with the 10CFR50 Technical Specifications for the plant, and the HI-STORM FW System Certificate of Compliance (CoC).

### Acceptance Tests and Maintenance

The acceptance criteria and maintenance program to be applied to the MPC are described in Chapter 10. The operational controls and limits to be applied to the MPC are discussed in

Chapter 13. 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 MPC is designed to be transportable in a HI-STAR overpack and is not required to be unloaded prior to shipment off-site. Decommissioning of the HI-STORM FW System is addressed in Section 2.4.

## 2.0.2 HI-STORM FW Overpack Design Criteria

### General

The HI-STORM FW overpack is engineered for a 60 year Design Life while satisfying the requirements of 10CFR72. The adequacy of the overpack to meet the required design life is discussed in Subsection 3.4.7. The design characteristics of the HI-STORM FW overpack are summarized in Subsection 1.2.1.

### Structural

The HI-STORM FW overpack includes both concrete and structural steel parts that are classified as important-to-safety.

The concrete material is defined as important-to-safety because of its shielding function. The primary function of the HI-STORM FW overpack concrete is shielding of the gamma and neutron radiation emitted by the spent nuclear fuel.

The HI-STORM FW overpack plain concrete is enclosed in steel inner and outer shells connected to each other by radial ribs, and top and bottom plates. As the HI-STORM FW overpack concrete is not reinforced, the structural analysis of the overpack only credits the compressive strength of the concrete in the analysis to provide an appropriate simulation of the accident conditions postulated in this FSAR. The technical requirements on testing and qualification of the HI-STORM FW overpack plain concrete are in Appendix 1.D of the HI-STORM 100 FSAR. Appendix 1.D is incorporated in this FSAR by reference.

There is no U.S. or international code that is sufficiently comprehensive to provide a completely prescriptive set of requirements for the design, manufacturing, and structural qualification of the overpack. The various sections of the ASME Codes, however, contain a broad range of specifications that can be assembled to provide a complete set of requirements for the design, analysis, shop manufacturing, and final field construction of the overpack. The portions or whole of the Codes and Standards that are invoked for the various elements of the overpack design, analysis, and manufacturing activities (viz., materials, fabrication and inspection) are summarized in Tables 1.2.6, and 1.2.7.

The ASME Boiler and Pressure Vessel Code (ASME Code) Section III, Subsection NF Class 3, [2.0.3], is the applicable code to determine stress limits for the load bearing components of the overpack when required by the acceptance criteria set down in this chapter. The material types used in the components of the HI-STORM FW System are listed in the licensing drawings.

ACI 318-05 [2.0.4] is the applicable reference code to establish the limits on unreinforced concrete (in the Closure Lid), which is subject to secondary structural loadings. Appendix 1.D contains the design, construction, and testing criteria applicable to the plain concrete in the overpack lid.

As mandated by 10CFR72.24(c)(3) and §72.44(d), Holtec International's quality assurance (QA) program requires all constituent parts of an SSC subject to NRC certification under 10CFR72 to be assigned an ITS category appropriate to its function in the control and confinement of radiation. The ITS designations (ITS or NITS) for the constituent parts of the HI-STORM FW System are provided in the licensing drawings. The QA categorization level (A, B, or C) for ITS parts is provided in Tables 2.0.1 through 2.0.8. A table exists for each licensing drawing and provides the QA level for the parts designated as ITS on the licensing drawings.

The excerpts from the codes, standards, and generally recognized industry publications invoked in this FSAR, supplemented by the commitments in Holtec's QA procedures, provide the necessary technical framework to ensure that the as-installed system would meet the intent of §72.24(c), §72.120(a) and §72.236(b). As required by Holtec's QA Program (discussed in Chapter 14), all operations on ITS components must be performed under QA validated written procedures and specifications that are in compliance with the governing citations of codes, standards, and practices set down in this FSAR.

The overpack is designed for all normal, off-normal, and design basis accident condition loadings, as defined in Section 2.2.

### Thermal

The thru-thickness temperature limits for the plain concrete in the overpack for long term and short term temperatures are in Table 2.2.3. 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 overpack is designed for extreme cold conditions, as discussed in Subsection 2.2.2. The brittle fracture assessment of structural steel materials used in the storage cask is considered in Section 3.1.

The overpack is designed to dissipate the maximum allowable heat load (shown in Tables 1.2.3 and 1.2.4) from the MPC. The thermal characteristics of the MPC stored inside the overpack are evaluated in Chapter 4.

## Shielding

The off-site dose for normal operating conditions to a real individual beyond the controlled area boundary is limited by 10CFR72.104(a) to a maximum of 25 mrem/year whole body, 75 mrem/year thyroid, and 25 mrem/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 on the ISFSI pad), the determination and comparison of ISFSI doses to this limit are necessarily site-specific. Dose rates for a single cask and a range of typical ISFSIs using the HI-STORM FW System are provided in Chapter 5. The determination of site-specific ISFSI dose rates at the site boundary and demonstration of compliance with regulatory limits is to be performed by the licensee in accordance with 10CFR72.212.

The overpack is designed to limit the calculated surface dose rates on the cask for all MPC designs as defined in Subsection 2.3.5. The overpack is also designed to maintain occupational exposures ALARA during MPC processing, in accordance with 10CFR20. The calculated overpack dose rates are determined in Section 5.1. These dose rates are used to perform a generic occupational exposure estimate for MPC operations and a site boundary dose assessment for a typical ISFSI, as described in Chapter 11.

## Confinement

The overpack does not perform any confinement function. Confinement during storage is provided by the MPC. The overpack provides physical protection and radiation shielding of the MPC contents during dry storage operations.

## Operations

There are no radioactive effluents that result from MPC operations after the MPC is sealed or during storage 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-STORM FW System are provided in Chapter 9. The licensee is required to develop detailed operating procedures based on Chapter 9 with due consideration of site-specific conditions including the applicable 10CFR50 technical specification requirements for the site, and the HI-STORM FW System CoC.

## Acceptance Tests and Maintenance

The acceptance criteria and maintenance program to be applied to the overpack are described in Chapter 10. The operational controls and limits to be applied to the overpack are contained in Chapter 13. 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-STORM FW System, including the overpack, are addressed in Section 2.4.

### 2.0.3 HI-TRAC VW Transfer Cask Design Criteria

#### General

The HI-TRAC VW transfer cask is engineered for a 60 year design life. The adequacy of the HI-TRAC VW to meet the above design life commitment is discussed in Section 3.4. The design characteristics of the HI-TRAC VW cask are presented in Subsection 1.2.1.

#### Structural

The HI-TRAC VW transfer cask includes both structural and non-structural radiation shielding components that are classified as important-to-safety. The structural steel components of the HI-TRAC VW are designed to meet the stress limits of Section III, Subsection NF, of the ASME Code for normal and off-normal storage conditions. The threaded anchor locations for lifting and handling of the transfer cask are designed in accordance with the requirements of NUREG-0612 and Regulatory Guide 3.61 for interfacing lift points.

The HI-TRAC VW transfer cask design is analyzed for all normal, off-normal, and design basis accident condition loadings, as defined in Section 2.2. Under accident conditions, the HI-TRAC VW transfer cask must protect the MPC from unacceptable deformation, provide continued shielding, and remain in a condition such that the MPC can be removed from it. The loads applicable to the HI-TRAC VW transfer cask are defined in Tables 2.2.6 and 2.2.13 and Table 3.1.1. The physical characteristics of each MPC for which the HI-TRAC VW is designed are presented in Subsection 1.2.1.

#### Thermal

The allowable temperatures for the HI-TRAC VW transfer cask structural steel components are based on the maximum temperature for material properties and allowable stress values provided in Section II of the ASME Code. The allowable temperatures for the structural steel and shielding components of the HI-TRAC VW are provided in Table 2.2.3. The HI-TRAC VW is designed for off-normal environmental cold conditions, as discussed in Subsection 2.2.2. The evaluation of the potential for brittle fracture in structural steel materials is presented in Section 3.1.

The HI-TRAC VW is designed and evaluated for the maximum heat load analyzed for storage operations. The maximum allowable temperature of water in the HI-TRAC jacket is a function of the internal pressure. To preclude over-pressurization of the water jacket due to boiling of the neutron shield liquid (water), the maximum temperature of the water is restricted to be less than the saturation temperature at the shell design pressure. Even though the analysis shows that the

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water jacket will not over-pressurize, a relief device is placed at the top of the water jacket shell. In addition, the water is precluded from freezing during off-normal cold conditions by limiting the minimum allowable operating temperature and by adding ethylene glycol. The thermal characteristics of the fuel for each MPC for which the transfer cask is designed are defined in Section 2.1. The working area ambient temperature limit for loading operations is limited in accordance with Table 2.2.2.

### Shielding

The HI-TRAC VW transfer cask provides shielding to maintain occupational exposures ALARA in accordance with 10CFR20, while also maintaining the maximum load on the plant's crane hook to below the rated capacity of the crane. As discussed in Subsection 1.2.1, the shielding in HI-TRAC VW is maximized within the constraint of the allowable weight at a plant site. The HI-TRAC VW calculated dose rates for a set of reference conditions are reported in Section 5.1. These dose rates are used to perform a generic occupational exposure estimate for MPC loading, closure, and transfer operations, as described in Chapter 11. A postulated HI-TRAC VW accident condition, which includes the loss of the liquid neutron shield (water), is also evaluated in Chapter 5.

The annular area between the MPC outer surface and the HI-TRAC VW inner surface can be isolated to minimize the potential for surface contamination of the MPC by spent fuel pool water during wet loading operations. The HI-TRAC VW surfaces expected to require decontamination are coated with a suitable coating. The maximum permissible surface contamination for the HI-TRAC VW is in accordance with plant-specific procedures and ALARA requirements (discussed in Chapter 11).

### Confinement

The HI-TRAC VW transfer cask does not perform any confinement function. The HI-TRAC VW provides physical protection and radiation shielding of the MPC contents during MPC loading, unloading, and transfer operations.

### Operations

There are no radioactive effluents that result from MPC transfer operations using HI-TRAC VW. Effluents generated during MPC loading and closure operations are handled by the plant's radwaste system and procedures.

Generic operating procedures for the HI-STORM FW System are provided in Chapter 9. The licensee will develop detailed operating procedures based on Chapter 9 along with plant-specific requirements including the Part 50 Technical Specification and SAR, and the HI-STORM FW System CoC.

### Acceptance Tests and Maintenance

The acceptance criteria and maintenance program to be applied to the HI-TRAC VW Transfer Cask are described in Chapter 10. The operational controls and limits to be applied to the HI-TRAC VW are contained in Chapter 13. Application of these requirements will assure that the HI-TRAC VW is fabricated, operated, and maintained in a manner that satisfies the design criteria given in this chapter.

### Decommissioning

Decommissioning considerations for the HI-STORM FW Systems, including the HI-TRAC VW transfer cask, are addressed in Section 2.4.

## 2.0.4 Principal Design Criteria for the ISFSI Pad

### 2.0.4.1 Design and Construction Criteria

In compliance with 10CFR72, Subpart F, “General Design Criteria”, the HI-STORM FW cask system is classified as “important-to-safety” (ITS). This FSAR explicitly recognizes the HI-STORM FW System as an assemblage of equipment containing numerous ITS components. The reinforced concrete pad, on which the cask is situated, however, is designated as a “not important to safety” (NITS) structure because of a lack of a physical connection between the cask and the pad.

Because the geological conditions vary widely across the United States, it is not possible to, *a priori*, define the detailed design of the ISFSI pad. Accordingly, in this FSAR, the limiting requirements on the design and installation of the pad are provided. The user of the HI-STORM FW System bears the responsibility to ensure that all requirements on the pad set forth in this FSAR are fulfilled by the pad design. Specifically, the ISFSI owner must ensure that:

- The pad design complies with the structural provisions of this FSAR.
- The material of construction of the pad (viz., the additives used in the pad concrete) are compatible with the ambient environment at the ISFSI site.
- Appropriate structural evaluations are performed pursuant to 10CFR72.212 to demonstrate that the pad is structurally competent to permit the cask to withstand the seismic and other credible inertial loadings at the site.

### 2.0.4.2 Load Combinations and Applicable Codes

Factored load combinations for ISFSI pad design are provided in NUREG-1536 [1.0.3]. The factored loads applicable to the pad design consist of dead weight of the cask, thermal gradient loads, impact loads arising from handling and accident events, external missiles, and bounding environmental phenomena (such as earthquakes, wind, tornado, and flood).

The factored load combinations presented in Table 3-1 of NUREG 1536 are reduced in number by eliminating loading types that are not germane or controlling in a HI-STORM ISFSI pad design. The applicable factored load combinations are accordingly adapted from the HI-STORM 100 FSAR and presented below.

a. Definitions

D = Dead load  
 L = Live load  
 T = Thermal load  
 E = DBE seismic load  
 $U_c$  = Reinforced concrete available strength

b. Load Combinations for the Concrete Pad

Normal Events

$$U_c > 1.4 D + 1.7 L$$

Off-Normal Events

$$U_c > 1.05 D + 1.275 (L+T)$$

Accidents

$$U_c > D + L + T + E$$

As an interfacing structure, the ISFSI pad and its underlying substrate must possess the structural strength to satisfy the above inequalities. As discussed in the HI-STAR 100 FSAR, thermal gradient loads are generally small; therefore, the Off-Normal Event does not generally provide a governing load combination.

Table 2.2.9 provides a reference set of parameters for the ISFSI pad and its foundation that are used solely as input to the non-mechanistic tipover analysis. Analyses in Chapter 3 show that this reference pad design does not violate the design criterion applicable to the non-mechanistic tipover of the HI-STORM FW storage system. The pad design may be customized to meet the requirements of a particular site, without performing a site-specific tipover analysis, provided that all ISFSI pad strength properties are less than or equal to the values in Table 2.2.9.

Applicable sections of industry codes such as ACI 318-05, "Building Code Requirements for Structural Concrete"; ACI 360R-92, "Design of Slabs on Grade"; ACI 302.1R, "Guide for Concrete Floor and Slab Construction"; and ACI 224R-90, "Control of Cracking in Concrete Structures" may be used in the design, structural evaluation, and construction of the concrete

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pad. However, load combinations in ACI 318-05 are not applicable to the ISFSI pad structural evaluation, and are replaced by the load combinations stated in subparagraph 2.0.4.2.b.

Table 2.0.1 – HI-STORM FW Assembly (Drawing # 6494)		
Item Number*	Part Name	ITS QA Safety Category
1	Assembly, Lid, HI-STORM Ø 113 B.C.	B
2	Lid-Stud	B
3	Heavy Hex Nut, 3 ¼” – 4 UNC	B
5	Plate, HI-STORM FW Heat Shield	B
6	Shielding, HI-STORM FW Body	B
8	Block, HI-STORM FW Cask Anchor	B
11	Plate, HI-STORM FW Body Base	B
15	Shell, HI-STORM FW Outer Shell	B
16	Shell, HI-STORM FW Inner Shell	B
17	Rib, HI-STORM FW Lifting Rib	B
18	Plate, HI-STORM FW Cask Body Top	B
20	Plate, Gamma Shield	C
21	Tube, MPC Guide	C
22	Tube, MPC Guide	C
23	Tube, MPC Guide	C
24	Closure Bolt	B

\*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

Table 2.0.2 – MPC-37 Enclosure Vessel (Drawing # 6505)		
Item Number*	Part Name	ITS QA Safety Category
1	Shell, Enclosure Vessel	A
2	Plate, Enclosure Vessel Base	A
3	Plate, Enclosure Vessel Lift Lug	C
4	Plate, Enclosure Vessel Upper Lid	A
5	Plate, Enclosure Vessel Lower Lid	B
6	Ring, Enclosure Vessel Closure	A
7	Block, Enclosure Vessel Vent/Drain Upper	B
8	Port, Enclosure Vessel Vent/Drain	C
9	Plug, Enclosure Vessel Vent /Drain	C
10	Block, Enclosure Vessel Lower Drain	C
12	Block, Enclosure Vessel Vent Shielding	C
13	Plate, Enclosure Vessel Vent/Drain Port Cover	A
16	Purge Tool Port Plug	C
21	Shim, Enclosure Vessel Type 1 PWR Fuel Basket	C
22	Shim, Enclosure Vessel Type 2 PWR Fuel Basket	C
23	Solid Shim	C

\*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

Table 2.0.3 – Assembly, MPC-37 Fuel Basket (Drawing # 6506)		
Item Number	Part Name	ITS QA Safety Category
1	Panel, Type 1 Cell Wall	A
2	Panel, Type 2 Cell Wall	A
3	Panel, Type 3 Cell Wall	A
4	Panel, Type 4 Cell Wall	A
5	Panel, Type 5 Cell Wall	A
6	Panel, Type 6 Cell Wall	A

Table 2.0.4 – Assembly, MPC-89 Fuel Basket (Drawing # 6507)		
Item Number	Part Name	ITS QA Safety Category
1	Panel, Type 1 Cell Wall	A
2	Panel, Type 2 Cell Wall	A
3	Panel, Type 3 Cell Wall	A
4	Panel, Type 4 Cell Wall	A
5	Panel, Type 5 Cell Wall	A
6	Panel, Type 6 Cell Wall	A
7	Panel, Type 7 Cell Wall	A
8	Panel, Type 8 Cell Wall	A

Table 2.0.5 – Assembly, Lid, HI-STORM Ø113 B.C. (Drawing # 6508)		
Item Number*	Part Name	ITS QA Safety Category
1	Plate, HI-STORM Lid Base	B
2	Plate, HI-STORM Lid Type 1 Round	B
3	Plate, HI-STORM Lid Type 2 Round	B
4	Plate, HI-STORM Lid Type 1 Ring	B
5	Plate, HI-STORM Lid Type 2 Ring	B
6	Plate, HI-STORM Lid Type 3 Ring	B
7	Plate, HI-STORM Lid Type 4 Ring	B
8	Plate, HI-STORM Lid Type 5 Ring	B
9	Plate, HI-STORM Lid Type 6 Ring	B
10	Plate, HI-STORM Lid Upper Shim	B
11	Plate, HI-STORM Lid Lower Shim	B
13	Gusset, HI-STORM Lid	B
16	Shielding, HI-STORM Lid Lower	B
17	Shielding, HI-STORM Lid Upper	B
18	Plate, Heat Shield	B
20	Block, HI-STORM Lid Lifting Anchor	B

\*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

Table 2.0.6 – MPC-89 Enclosure Vessel (Drawing # 6512)		
Item Number*	Part Name	ITS QA Safety Category
1	Shell, Enclosure Vessel	A
2	Plate, Enclosure Vessel Base	A
3	Plate, Enclosure Vessel Lift Lug	C
4	Plate, Enclosure Vessel Upper Lid	A
5	Plate, Enclosure Vessel Lower Lid	B
6	Ring, Enclosure Vessel Closure	A
7	Block, Enclosure Vessel Vent/Drain Upper	B
8	Port, Enclosure Vessel Vent/Drain	C
9	Plug, Enclosure Vessel Vent/Drain	C
10	Block, Enclosure Vessel Lower Drain	C
12	Block, Enclosure Vessel Vent Shielding	C
13	Plate, Enclosure Vessel Vent/Drain Port Cover	A
16	Purge Tool Port Plug	C
21	Shim, Enclosure Vessel Type 1 BWR Fuel Basket	C
22	Shim, Enclosure Vessel Type 2 BWR Fuel Basket	C
23	Shim, Enclosure Vessel Type 3 BWR Fuel Basket	C
24	Solid Shim	C

\*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

Table 2.0.7 – HI-TRAC VW – MPC-37 (Drawing # 6514)		
Item Number*	Part Name	ITS QA Safety Category
1	Flange, Bottom	B
3	Hex Bolt, 2-4 ½ UNC X 6” LG.	B
4	Shell, Inner	B
5	Shielding, Gamma	B
6	Flange, Top	A
7	Shell, Water Jacket	B
10	Pipe, Bolt Recess	B
11	Cap, Bolt Recess	B
12	Bottom Lid	B
13	Shell, Outer	B
14	Rib, Extended	B
15	Rib, Short	B

\*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.



Table 2.0.8 – HI-TRAC VW – MPC-89 (Drawing # 6799)		
Item Number*	Part Name	ITS QA Safety Category
1	Flange, Bottom	B
3	Hex Bolt, 2-4 ½ UNC X 6” LG.	B
4	Shell, Inner	B
5	Shielding, Gamma	B
6	Flange, Top	A
7	Shell, Water Jacket	B
10	Pipe, Bolt Recess	B
11	Cap, Bolt Recess	B
12	Bottom Lid	B
13	Shell, Outer	B
14	Rib, Extended	B
15	Rib, Short	B

\*Item Numbers are non-consecutive because they are consistent with Parts List on drawing.

## 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-STORM FW System is to ensure that all SNF discharged from the U.S. reactors and not yet loaded into dry storage systems can be stored in a HI-STORM FW MPC. Publications such as references [2.1.1] and [2.1.2] provide a comprehensive description of fuel discharged from U.S. reactors.

The cell openings in the fuel baskets have been sized to accommodate BWR and PWR assemblies. The cavity length of the MPC will be determined for a specific site to accord with the fuel assembly length used at that site, including non-fuel hardware and damaged fuel containers, as applicable.

Table 2.1.1 summarizes the authorized contents for the HI-STORM FW System. Tables 2.1.2 and 2.1.3, which are referenced in Table 2.1.1, provide the fuel characteristics of all groups of fuel assembly types determined to be acceptable for storage in the HI-STORM FW System. Any fuel assembly that has fuel characteristics within the range of Tables 2.1.2 and 2.1.3 and meets the other limits specified in Table 2.1.1 is acceptable for storage in the HI-STORM FW System. The groups of fuel assembly types presented in Tables 2.1.2 and 2.1.3 are defined as “array/classes” as described in further detail in Chapter 6. Table 2.1.4 lists the BWR and PWR fuel assembly designs which are found to govern for three qualification criteria, namely reactivity, shielding, and thermal, or that are used as reference assembly design is those analyses. Additional information on the design basis fuel definition is presented in the following subsections.

### 2.1.2 Undamaged SNF Specifications

Undamaged fuel is defined in the Glossary.

### 2.1.3 Damaged SNF and Fuel Debris Specifications

Damaged fuel and fuel debris are defined in the Glossary.

Damaged fuel assemblies and fuel debris will be loaded into damaged fuel containers (DFCs) (Figure 2.1.6) that have mesh screens on the top and bottom. The DFC will have a removable lid to allow the fuel assembly to be inserted. In storage, the lid will be latched in place. DFC's used to move fuel assemblies will be designed for lifting with either the lid installed or with a separate handling lid. DFC's used to handle fuel and the associated lifting tools will be designed in accordance with the requirements of NUREG-0612. The DFC will be fabricated from structural aluminum or stainless steel. The appropriate structural, thermal, shielding, criticality, and confinement evaluations have been performed to account for damaged fuel and fuel debris and are described in their respective chapters that follow. The limiting design characteristics for

damaged fuel assemblies and restrictions on the number and location of damaged fuel containers authorized for loading in each MPC model are provided in this chapter.

## 2.1.4 Structural Parameters for Design Basis SNF

The main physical parameters of an SNF assembly applicable to the structural evaluation are the fuel assembly length, cross sectional dimensions, and weight. These parameters, which define the mechanical and structural design, are specified in Subsection 2.1.8. An appropriate axial clearance is provided to prevent interference due to the irradiation and thermal growth of the fuel assemblies.

## 2.1.5 Thermal Parameters for Design Basis SNF

The principal thermal design parameter for the stored fuel is the fuel's peak cladding temperature (PCT) which is a function of the maximum decay heat per assembly and the decay heat removal capabilities of the HI-STORM FW System.

To ensure the permissible PCT limits are not exceeded, Subsection 1.2 specifies the maximum allowable decay heat per assembly for each MPC model in the three-region configuration (see also Table 1.2.3 and 1.2.4).

The fuel cladding temperature is also affected by the heat transfer characteristics of the fuel assemblies. The design basis fuel assembly for thermal calculations for both PWR and BWR fuel is provided in Table 2.1.4.

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 references [2.1.3] and [2.1.4] are utilized and summarized in Table 2.1.5 and Figures 2.1.3 and 2.1.4. 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-STORM FW System.

## 2.1.6 Radiological Parameters for Design Basis SNF

The principal radiological design criteria for the HI-STORM FW System are the 10CFR72 §104 and §106 operator-controlled boundary dose rate limits, and the requirement to maintain operational dose rates as low as reasonably achievable (ALARA). The radiation dose is directly affected by the gamma and neutron source terms of the assembly, which is a function of the assembly type, and the burnup, enrichment and cooling time of the assemblies. Dose rates are further directly affected by the size and arrangement of the ISFSI, and the specifics of the loading operations. All these parameters are site-dependent, and the compliance with the regulatory dose rate requirements are performed in site-specific calculations. The evaluations here are therefore performed with reference fuel assemblies, and with parameters that result in

reasonably conservative dose rates. The reference assemblies given in Table 1.0.4 are the predominant assemblies used in the industry.

The design basis dose rates can be met by a variety of burnup levels and cooling times. Table 2.1.1 provides the acceptable ranges of burnup, enrichment and cooling time for all of the authorized fuel assembly array/classes. Table 2.1.5 and Figures 2.1.3 and 2.1.4 provide 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-STORM FW System.

Non-fuel hardware, as defined in the Glossary, has been evaluated and is also authorized for storage in the PWR MPCs as specified in Table 2.1.1.

### 2.1.7 Criticality Parameters for Design Basis SNF

The criticality analyses for the MPC-37 are performed with credit taken for soluble boron in the MPC water during wet loading and unloading operations. Table 2.1.6 provides the required soluble boron concentrations for this MPC.

### 2.1.8 Summary of Authorized Contents

Tables 2.1.1 through 2.1.3 specify the limits for spent fuel and non-fuel hardware authorized for storage in the HI-STORM FW System. The limits in these tables are derived from the safety analyses described in the following chapters of this FSAR.

Table 2.1.1		
MATERIAL TO BE STORED		
PARAMETER	VALUE	
	MPC-37	MPC-89
Fuel Type	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, and fuel debris meeting the limits in Table 2.1.2 for the applicable array/class.	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, with or without channels, fuel debris meeting the limits in Table 2.1.3 for the applicable array/class.
Cladding Type	ZR (see Glossary for definition)	ZR (see Glossary for definition)
Maximum Initial Rod Enrichment	Depending on soluble boron levels and assembly array/class as specified in Table 2.1.6	$\leq 5.0$ wt. % U-235
Post-irradiation cooling time and average burnup per assembly	Minimum Cooling Time: 3 years  Maximum Assembly Average Burnup: 68.2 GWd/mtU	Minimum Cooling Time: 3 years  Maximum Assembly Average Burnup: 65 GWd/mtU
Non-fuel hardware post-irradiation cooling time and burnup	Minimum Cooling Time: 3 years  Maximum Burnup <sup>†</sup> : - BPRAs, WABAs and vibration suppressors: 60 GWd/mtU - TPDs, NSAs, APSRs, RCCAs, CRAs, CEAs, water displacement guide tube plugs and orifice rod assemblies: 630 GWd/mtU - ITTRs: not applicable	N/A
Decay heat per fuel storage location	Regionalized Loading: See Table 1.2.3	Regionalized Loading: See Table 1.2.4

<sup>†</sup> Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation. Burnup not applicable for ITTRs since installed post-irradiation.

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Table 2.1.1 (continued)		
MATERIAL TO BE STORED		
PARAMETER	VALUE	
	MPC-37	MPC-89
Fuel Assembly Nominal Length (in.)	Minimum: (1) All except 15x15I <sup>‡</sup> : 157 (with NFH); (2) 15x15I: 149 (with NFH) § Reference: 167.2 (with NFH) Maximum: 199.2 (with NFH and DFC)	Minimum: 171 Reference: 176.5 Maximum: 181.5 (with DFC)
Fuel Assembly Width (in.)	≤ 8.54 (nominal design)	≤ 5.95 (nominal design)
Fuel Assembly Weight (lb)	Reference: 1600 (without NFH) 1750 (with NFH), 1850 (with NFH and DFC)  Maximum: 2050 (including NFH and DFC)	Reference: 750 (without DFC), 850 (with DFC)  Maximum: 850 (including DFC)

<sup>‡</sup> See Table 2.1.2 for 15x15I fuel assembly array/class characteristics.

§ Minimum nominal fuel assembly length for 15x15I fuel assembly array/class is 149". The unique design of 15x15I fuel requires a 1" nominal fuel shim to properly support the assembly. Therefore the minimum MPC cavity height for 15x15I fuel is based on 150" fuel length.

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Table 2.1.1 (continued)		
MATERIAL TO BE STORED		
PARAMETER	VALUE	
	MPC-37	MPC-89
Other Limitations	<ul style="list-style-type: none"> <li>▪ Quantity is limited to 37 undamaged ZR clad PWR fuel assemblies with or without non-fuel hardware. Up to 12 damaged fuel containers containing PWR damaged fuel and/or fuel debris may be stored in the locations denoted in Figure 2.1.1 with the remaining basket cells containing undamaged ZR fuel assemblies, up to a total of 37.</li> <li>▪ One NSA.</li> <li>▪ Up to 30 BPRAs.</li> <li>▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location.</li> <li>▪ CRAs, RCCAs, CEAs, NSAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations specified in Figure 2.1.5.</li> </ul>	<ul style="list-style-type: none"> <li>▪ Quantity is limited to 89 undamaged ZR clad BWR fuel assemblies. Up to 16 damaged fuel containers containing BWR damaged fuel and/or fuel debris may be stored in locations denoted in Figure 2.1.2 with the remaining basket cells containing undamaged ZR fuel assemblies, up to a total of 89.</li> </ul>

Table 2.1.2					
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)					
Fuel Assembly Array/ Class	14x14 A	14x14 B	14x14 C	15x15 B	15x15 C
No. of Fuel Rod Locations	179	179	176	204	204
Fuel Clad O.D. (in.)	$\geq 0.400$	$\geq 0.417$	$\geq 0.440$	$\geq 0.420$	$\geq 0.417$
Fuel Clad I.D. (in.)	$\leq 0.3514$	$\leq 0.374$	$\leq 0.3880$	$\leq 0.3736$	$\leq 0.3640$
Fuel Pellet Dia. (in.) (Note 3)	$\leq 0.3444$	$\leq 0.367$	$\leq 0.3805$	$\leq 0.3671$	$\leq 0.3570$
Fuel Rod Pitch (in.)	$\leq 0.556$	$\leq 0.566$	$\leq 0.580$	$\leq 0.563$	$\leq 0.563$
Active Fuel Length (in.)	$\leq 150$	$\leq 150$	$\leq 150$	$\leq 150$	$\leq 150$
No. of Guide and/or Instrument Tubes	17	17	5 (Note 2)	21	21
Guide/Instrument Tube Thickness (in.)	$\geq 0.017$	$\geq 0.017$	$\geq 0.038$	$\geq 0.015$	$\geq 0.0165$



Table 2.1.2 (continued)					
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)					
Fuel Assembly Array/Class	15x15 D	15x15 E	15x15 F	15x15 H	15x15 I
No. of Fuel Rod Locations	208	208	208	208	216
Fuel Clad O.D. (in.)	$\geq 0.430$	$\geq 0.428$	$\geq 0.428$	$\geq 0.414$	$\geq 0.413$
Fuel Clad I.D. (in.)	$\leq 0.3800$	$\leq 0.3790$	$\leq 0.3820$	$\leq 0.3700$	$\leq 0.3670$
Fuel Pellet Dia. (in.) (Note 3)	$\leq 0.3735$	$\leq 0.3707$	$\leq 0.3742$	$\leq 0.3622$	$\leq 0.3600$
Fuel Rod Pitch (in.)	$\leq 0.568$	$\leq 0.568$	$\leq 0.568$	$\leq 0.568$	$\leq 0.550$
Active Fuel Length (in.)	$\leq 150$	$\leq 150$	$\leq 150$	$\leq 150$	$\leq 150$
No. of Guide and/or Instrument Tubes	17	17	17	17	9 (Note 4)
Guide/Instrument Tube Thickness (in.)	$\geq 0.0150$	$\geq 0.0140$	$\geq 0.0140$	$\geq 0.0140$	$\geq 0.0140$

Table 2.1.2 (continued)						
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)						
Fuel Assembly Array and Class	16x16 A	17x17A	17x17 B	17x17 C	17x17 D	17x17 E
No. of Fuel Rod Locations	236	264	264	264	264	265
Fuel Clad O.D. (in.)	$\geq 0.382$	$\geq 0.360$	$\geq 0.372$	$\geq 0.377$	$\geq 0.372$	$\geq 0.372$
Fuel Clad I.D. (in.)	$\leq 0.3350$	$\leq 0.3150$	$\leq 0.3310$	$\leq 0.3330$	$\leq 0.3310$	$\leq 0.3310$
Fuel Pellet Dia. (in.) (Note 3)	$\leq 0.3255$	$\leq 0.3088$	$\leq 0.3232$	$\leq 0.3252$	$\leq 0.3232$	$\leq 0.3232$
Fuel Rod Pitch (in.)	$\leq 0.506$	$\leq 0.496$	$\leq 0.496$	$\leq 0.502$	$\leq 0.496$	$\leq 0.496$
Active Fuel length (in.)	$\leq 150$	$\leq 150$	$\leq 150$	$\leq 150$	$\leq 170$	$\leq 170$
No. of Guide and/or Instrument Tubes	5 (Note 2)	25	25	25	25	24
Guide/Instrument Tube Thickness (in.)	$\geq 0.0350$	$\geq 0.016$	$\geq 0.014$	$\geq 0.020$	$\geq 0.014$	$\geq 0.014$

## 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. Each guide tube replaces four fuel rods.
3. Annular fuel pellets are allowed in the top and bottom 12" of the active fuel length.
4. One Instrument Tube and eight Guide Bars (Solid ZR).

Table 2.1.3					
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)					
Fuel Assembly Array and Class	7x7 B	8x8 B	8x8 C	8x8 D	8x8 E
Maximum Planar-Average Initial Enrichment (wt.% <sup>235</sup> U) (Note 14)	≤ 4.8	≤ 4.8	≤ 4.8	≤ 4.8	≤ 4.8
No. of Fuel Rod Locations	49	63 or 64	62	60 or 61	59
Fuel Clad O.D. (in.)	≥ 0.5630	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930
Fuel Clad I.D. (in.)	≤ 0.4990	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250
Fuel Pellet Dia. (in.)	≤ 0.4910	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160
Fuel Rod Pitch (in.)	≤ 0.738	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 10)	0	1 or 0	2	1 - 4 (Note 6)	5
Water Rod Thickness (in.)	N/A	≥ 0.034	> 0.00	> 0.00	≥ 0.034
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100

Table 2.1.3 (continued)					
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)					
Fuel Assembly Array and Class	8x8F	9x9 A	9x9 B	9x9 C	9x9 D
Maximum Planar-Average Initial Enrichment (wt.% <sup>235</sup> U) (Note 14)	≤ 4.5 (Note 12)	≤ 4.8	≤ 4.8	≤ 4.8	≤ 4.8
No. of Fuel Rod Locations	64	74/66 (Note 4)	72	80	79
Fuel Clad O.D. (in.)	≥ 0.4576	≥ 0.4400	≥ 0.4330	≥ 0.4230	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3996	≤ 0.3840	≤ 0.3810	≤ 0.3640	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3913	≤ 0.3760	≤ 0.3740	≤ 0.3565	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.609	≤ 0.566	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 10)	N/A (Note 2)	2	1 (Note 5)	1	2
Water Rod Thickness (in.)	≥ 0.0315	> 0.00	> 0.00	≥ 0.020	≥ 0.0300
Channel Thickness (in.)	≤ 0.055	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.100

Table 2.1.3 (continued)					
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)					
Fuel Assembly Array and Class	9x9 E (Note 3)	9x9 F (Note 3)	9x9 G	10x10 A	10x10 B
Maximum Planar-Average Initial Enrichment (wt.% <sup>235</sup> U) (Note 14)	≤ 4.5 (Note 12)	≤ 4.5 (Note 12)	≤ 4.8	≤ 4.8	≤ 4.8
No. of Fuel Rod Locations	76	76	72	92/78 (Note 7)	91/83 (Note 8)
Fuel Clad O.D. (in.)	≥ 0.4170	≥ 0.4430	≥ 0.4240	≥ 0.4040	≥ 0.3957
Fuel Clad I.D. (in.)	≤ 0.3640	≤ 0.3860	≤ 0.3640	≤ 0.3520	≤ 0.3480
Fuel Pellet Dia. (in.)	≤ 0.3530	≤ 0.3745	≤ 0.3565	≤ 0.3455	≤ 0.3420
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.510	≤ 0.510
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 10)	5	5	1 (Note 5)	2	1 (Note 5)
Water Rod Thickness (in.)	≥ 0.0120	≥ 0.0120	≥ 0.0320	≥ 0.030	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.120

Table 2.1.3 (continued)			
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)			
Fuel Assembly Array and Class	10x10 C	10x10 F	10x10 G
Maximum Planar-Average Initial Enrichment (wt.% <sup>235</sup> U) (Note 14)	≤ 4.8	≤ 4.7 (Note 13)	≤ 4.6 (Note 12)
No. of Fuel Rod Locations	96	92/78 (Note 7)	96/84
Fuel Clad O.D. (in.)	≥ 0.3780	≥ 0.4035	≥ 0.387
Fuel Clad I.D. (in.)	≤ 0.3294	≤ 0.3570	≤ 0.340
Fuel Pellet Dia. (in.)	≤ 0.3224	≤ 0.3500	≤ 0.334
Fuel Rod Pitch (in.)	≤ 0.488	≤ 0.510	≤ 0.512
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 10)	5 (Note 9)	2	5 (Note 9)
Water Rod Thickness (in.)	≥ 0.031	≥ 0.030	≥ 0.031
Channel Thickness (in.)	≤ 0.055	≤ 0.120	≤ 0.060

Table 2.1.3 (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. This assembly is known as "QUAD+." It has four rectangular water cross segments dividing the assembly into four quadrants.
3. 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
4. This assembly class contains 74 total rods; 66 full length rods and 8 partial length rods.
5. Square, replacing nine fuel rods.
6. Variable.
7. This assembly contains 92 total fuel rods; 78 full length rods and 14 partial length rods.
8. This assembly class contains 91 total fuel rods; 83 full length rods and 8 partial length rods.
9. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.
10. These rods may also be sealed at both ends and contain ZR material in lieu of water.
11. Not Used
12. When loading fuel assemblies classified as damaged fuel assemblies, all assemblies in the MPC are limited to 4.0 wt.% U-235.
13. When loading fuel assemblies classified as damaged fuel assemblies, all assemblies in the MPC are limited to 4.6 wt.% U-235.
14. In accordance with the definition of undamaged fuel assembly, certain assemblies may be limited to 3.3 wt.% U-235. When loading these fuel assemblies, all assemblies in the MPC are limited to 3.3 wt.% U-235.

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Table 2.1.4		
DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION		
<b>Criterion</b>	<b>BWR</b>	<b>PWR</b>
Reactivity/Criticality	GE-12/14 10x10 (Array/Class 10x10A)	Westinghouse 17x17 OFA (Array/Class 17x17B)
Shielding	GE-12/14 10x10	Westinghouse 17x17 OFA
Thermal-Hydraulic	GE-12/14 10x10	Westinghouse 17x17 OFA



Table 2.1.5  
NORMALIZED DISTRIBUTION BASED ON BURNUP PROFILE

<b>PWR DISTRIBUTION<sup>1</sup></b>		
<b>Interval</b>	<b>Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)</b>	<b>Normalized Distribution</b>
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
<b>BWR DISTRIBUTION<sup>2</sup></b>		
<b>Interval</b>	<b>Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)</b>	<b>Normalized Distribution</b>
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

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<sup>1</sup> Reference 2.1.7

<sup>2</sup> Reference 2.1.8

Table 2.1.6

## Soluble Boron Requirements for MPC-37 Wet Loading and Unloading Operations

Array/Class	All Undamaged Fuel Assemblies		One or More Damaged Fuel Assemblies and/or Fuel Debris	
	Maximum Initial Enrichment $\leq 4.0 \text{ wt}\% \text{ }^{235}\text{U}$ (ppmb)	Maximum Initial Enrichment $5.0 \text{ wt}\% \text{ }^{235}\text{U}$ (ppmb)	Maximum Initial Enrichment $\leq 4.0 \text{ wt}\% \text{ }^{235}\text{U}$ (ppmb)	Maximum Initial Enrichment $5.0 \text{ wt}\% \text{ }^{235}\text{U}$ (ppmb)
All 14x14 and 16x16A	1,000	1,500	1,300	1,800
All 15x15 and 17x17	1,500	2,000	1,800	2,300

## Note:

1. For maximum initial enrichments between  $4.0 \text{ wt}\%$  and  $5.0 \text{ wt}\% \text{ }^{235}\text{U}$ , the minimum soluble boron concentration may be determined by linear interpolation between the minimum soluble boron concentrations at  $4.0 \text{ wt}\%$  and  $5.0 \text{ wt}\% \text{ }^{235}\text{U}$ .

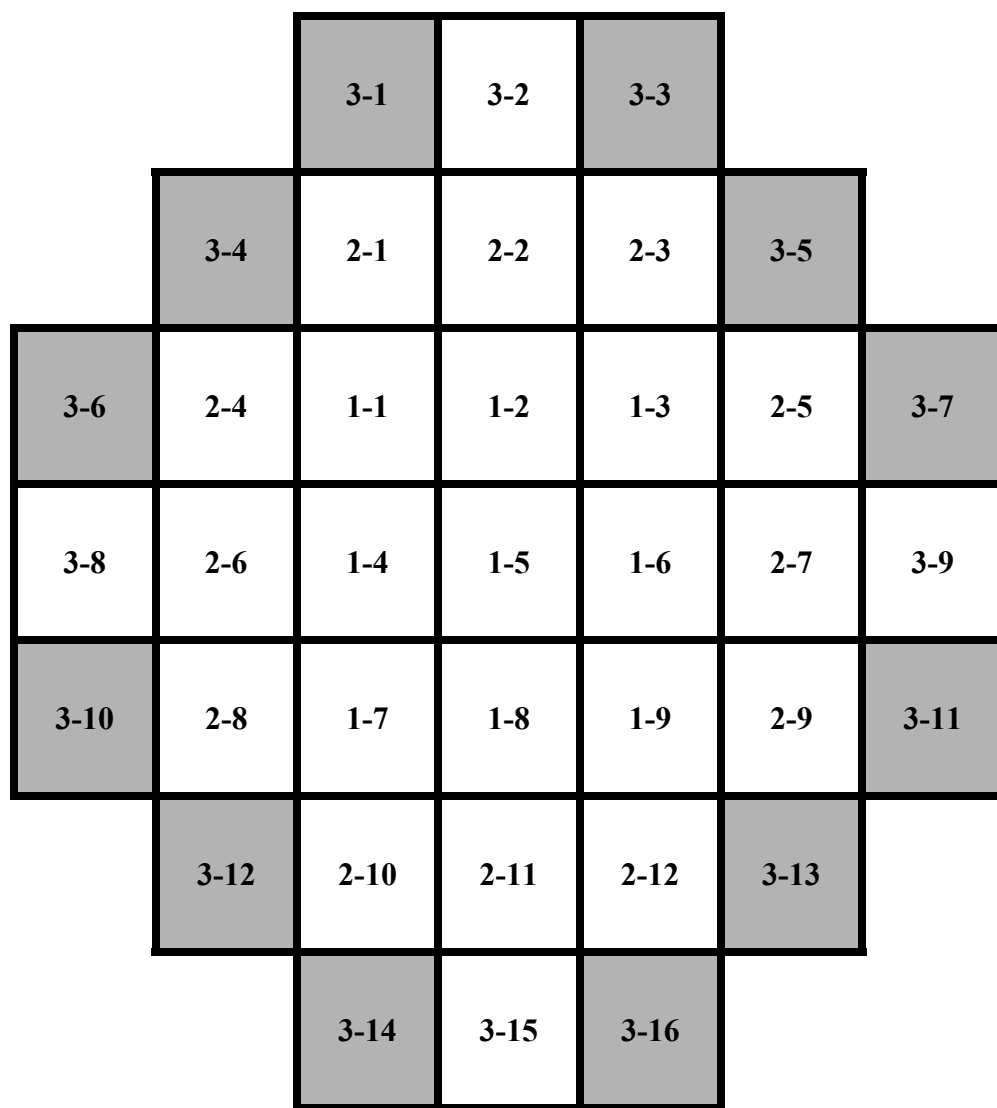


Figure 2.1.1 Location of DFCs for Damaged Fuel or Fuel Debris  
in the MPC-37(Shaded Cells)

				3-1	3-2	3-3				
		3-4	3-5	3-6	2-1	3-7	3-8	3-9		
	3-10	3-11	2-2	2-3	2-4	2-5	2-6	3-12	3-13	
	3-14	2-7	2-8	2-9	2-10	2-11	2-12	2-13	3-15	
3-16	3-17	2-14	2-15	1-1	1-2	1-3	2-16	2-17	3-18	3-19
3-20	2-18	2-19	2-20	1-4	1-5	1-6	2-21	2-22	2-23	3-21
3-22	3-23	2-24	2-25	1-7	1-8	1-9	2-26	2-27	3-24	3-25
	3-26	2-28	2-29	2-30	2-31	2-32	2-33	2-34	3-27	
	3-28	3-29	2-35	2-36	2-37	2-38	2-39	3-30	3-31	
		3-32	3-33	3-34	2-40	3-35	3-36	3-37		
				3-38	3-39	3-40				

Figure 2.1.2 Location of DFCs for Damaged Fuel or Fuel Debris  
in the MPC-89 (Shaded Cells)

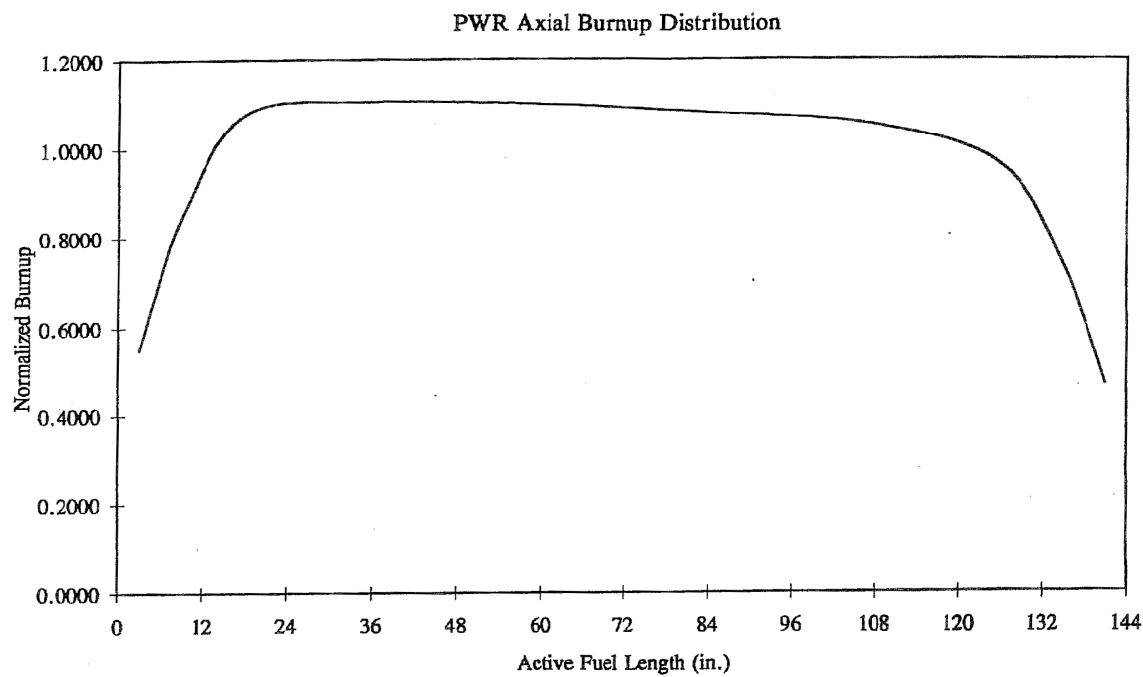


Figure 2.1.3 PWR Axial Burnup Profile with Normalized Distribution

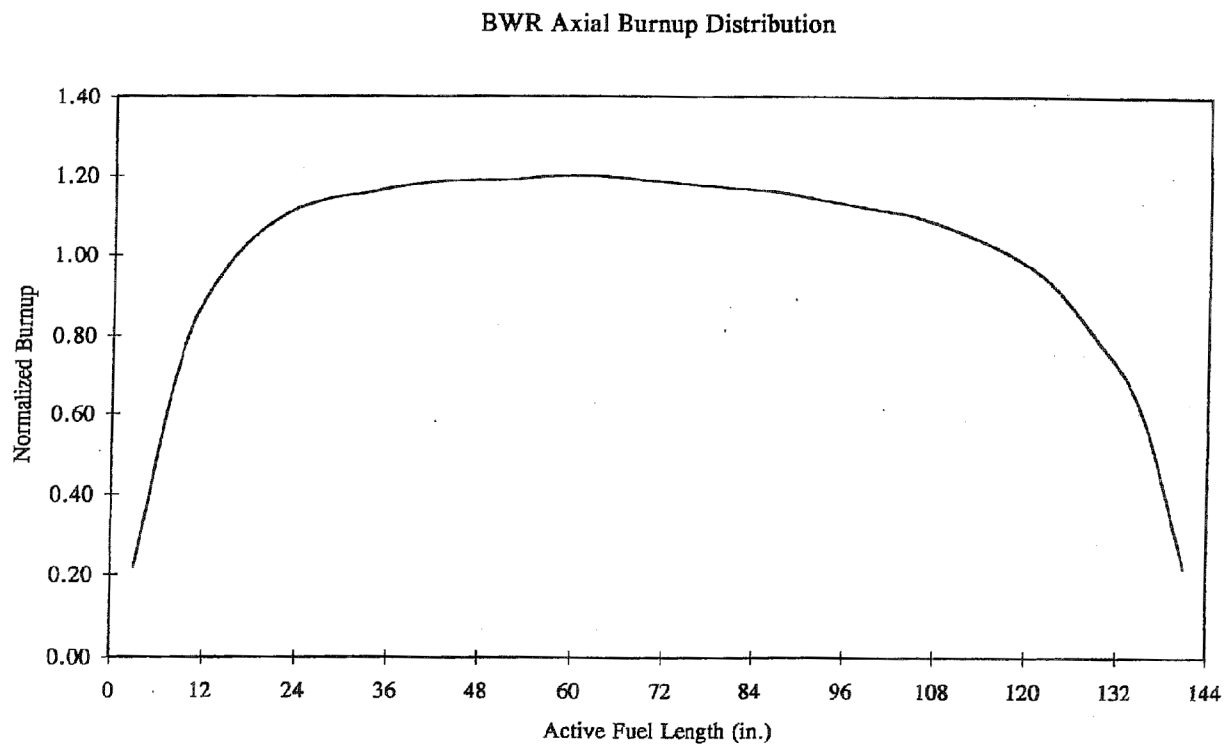


Figure 2.1.4 BWR Axial Burnup Profile with Normalized Distribution

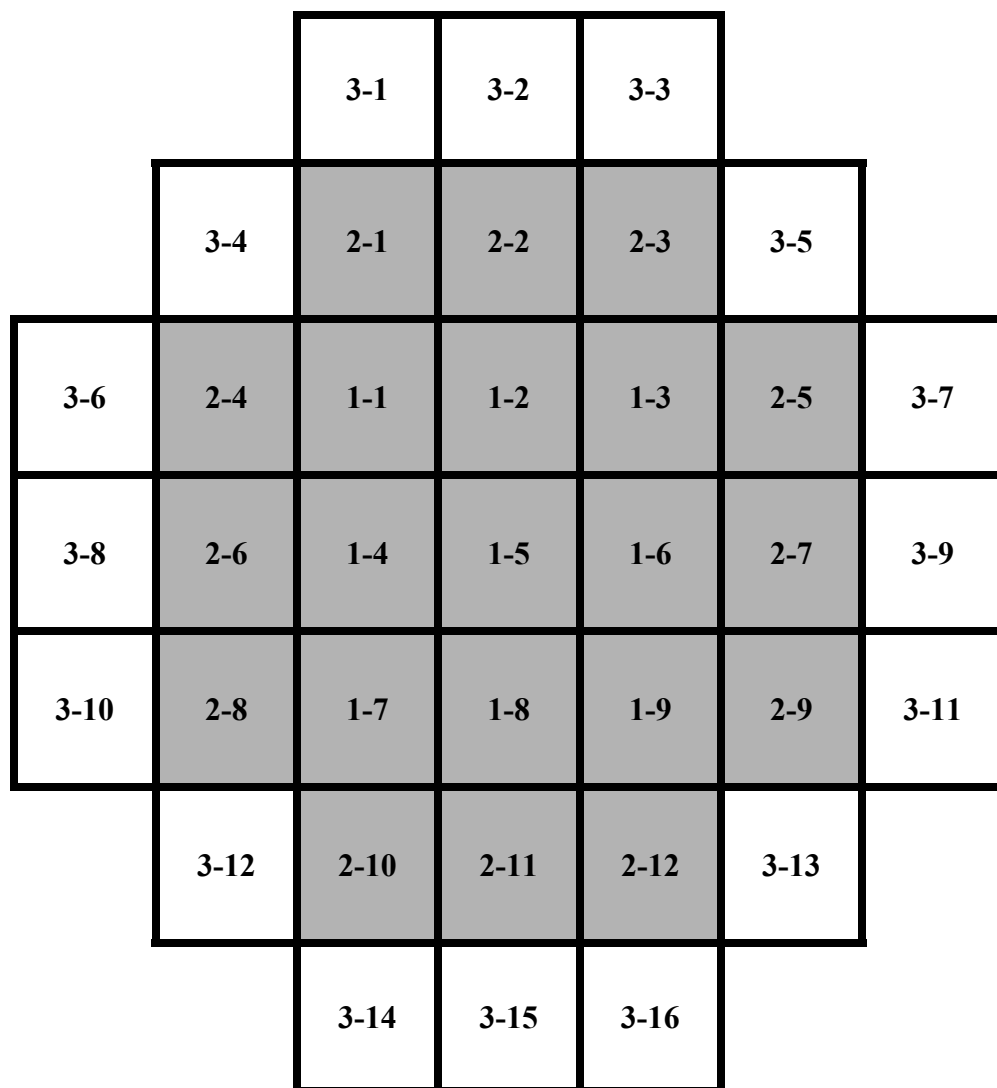


Figure 2.1.5: Location of NSAs, APSRs, RCCAs, CEAs, and CRAs in the MPC-37  
(Shaded Cells)

**Withheld in Accordance with 10 CFR 2.390**

Figure 2.1.6: Damaged Fuel Container (Typical)



## 2.2 HI-STORM FW DESIGN LOADINGS

The HI-STORM FW 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 and 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 and accident conditions include events that are postulated because their consideration establishes a conservative design basis.

Normal condition loads act in combination with all other loads (off-normal or environmental phenomena/accident). Off-normal condition loads and environmental phenomena and accident condition loads are not applied in combination. However, loads that occur as a result of the same phenomena are applied simultaneously. For example, the tornado winds 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. The following conditions of storage and associated loads are identified:

- i. Normal (Long-Term Storage) Condition: Dead Weight, Handling, Pressure, Temperature, Snow.
- ii. Off-Normal Condition: Pressure, Temperature, Leakage of One Seal, Partial Blockage of Air Inlets.
- iii. Accident Condition: Handling Accident, Non-Mechanistic Tip-Over, Fire, Partial Blockage of MPC Basket Flow Holes, Tornado, Flood, Earthquake, Fuel Rod Rupture, Confinement Boundary Leakage, Explosion, Lightning, Burial Under Debris, 100% Blockage of Air Inlets, Extreme Environmental Temperature.
- iv. Short-Term Operations: This loading condition is defined to accord with ISG-11, Revision 3 [2.0.1] guidance. 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, and on-site handling of a loaded HI-TRAC VW transfer cask.

Each of these conditions and the applicable loads are identified herein with their applicable design criteria. A design criterion is deemed to be satisfied if the allowable limits for the specific loading conditions are not exceeded.

## 2.2.1 Loadings Applicable to Normal Conditions of Storage

### a. Dead Weight

The HI-STORM FW System must withstand the static loads due to the weights of each of its components, including the weight of the HI-TRAC VW with the loaded MPC stacked on top the storage overpack during the MPC transfer.

### b. Handling Evolutions

The HI-STORM FW System must withstand loads experienced during routine handling. Normal handling includes:

- i. Vertical lifting and transfer to the ISFSI of the HI-STORM FW overpack containing a loaded MPC.
- ii. Vertical lifting and handling of the HI-TRAC VW transfer cask containing a loaded MPC.
- iii. Lifting of a loaded MPC.

The dead load of the lifted component is increased by 15% in the stress qualification analyses (to meet ANSI N14.6 guidance) to account for dynamic effects from lifting operations as suggested in CMAA #70 [2.2.1].

Handling operations of the loaded HI-TRAC VW transfer cask or HI-STORM FW overpack are limited to working area ambient temperatures specified in Table 2.2.2. This limitation is specified to ensure a sufficient safety margin against brittle fracture during handling operations.

Table 2.2.6 summarizes the analyses required to qualify all threaded anchor locations in the HI-STORM FW System.

### c. Pressure

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 an assumed 1% of the fuel rods ruptured with 100% of the fill gas and 30% of the significant radioactive gases (e.g.,  $H^3$ , Kr, and Xe) released in accordance with NUREG-1536.

For the storage of damaged fuel assemblies or fuel debris in a damaged fuel container (DFC), it shall be conservatively assumed that 100% of the fuel rods are ruptured with 100% of the rod fill gas and 30% of the significant radioactive gases (e.g.,  $H^3$ , Kr, and Xe) liberated. For PWR assemblies stored with non-fuel hardware, 100% of the gases in the non-fuel hardware (e.g.,

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BPRAs) shall be assumed to be released. The accident condition design pressure shall envelop the case of 100% of the fuel rods ruptured.

The MPC internal pressure under the normal condition of storage must remain below the design pressure specified in Table 2.2.1.

The MPC external pressure is a function of environmental conditions, which may produce a pressure loading. The normal condition external design pressure is specified in Table 2.2.1.

The HI-STORM FW overpack is not capable of retaining internal pressure due to its open design, and therefore no analysis is required or provided for the overpack internal pressure.

The HI-TRAC VW transfer cask is not capable of retaining internal pressure due to its open design. Therefore, no analysis is required for the internal pressure loading in HI-TRAC VW transfer cask. However, the HI-TRAC VW transfer cask water jacket may experience an internal vapor pressure due to the heat-up of the water contained in the water jacket. Analysis is performed in Chapter 3 of this report to demonstrate that the water jacket can withstand the design pressure in Table 2.2.1 without a structural failure and that the water jacket design pressure will not be exceeded. To provide an additional layer of safety, a pressure relief device is used to ensure that the water jacket design pressure will not be exceeded.

d. Environmental Temperatures and Pressures

To evaluate the long-term effects of ambient temperatures on the HI-STORM FW System, an upper bound value on the annual average ambient temperature for the continental United States is used. The annual average temperature is termed the normal ambient temperature for storage. The normal ambient temperature specified in Table 2.2.2 is bounding for all reactor sites in the contiguous United States. The normal ambient temperature set forth in Table 2.2.2 is intended to ensure that it is greater than the annual average of ambient temperature at any location in the continental United States. In the northern region of the U.S., the design basis normal ambient 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° F in Docket 72-1007) by utilizing national meteorological data.

Likewise, within the thermal analysis, a conservatively assumed soil temperature of the value specified in Table 2.2.2 is utilized to bound the annual average soil temperatures for the continental United States. The 1987 ASHRAE Handbook (HVAC Systems and Applications) reports average earth temperatures, from 0 to 10 feet below grade, throughout the continental United States. The highest reported annual average value for the continental United States is 77°F for Key West, Florida. Therefore, this value is specified in Table 2.2.2 as the bounding soil temperature.

Confirmation of the site-specific annual average ambient temperature and soil temperature is to be performed by the licensee, in accordance with 10CFR72.212. Insolation based on 10CFR71.71 input averaged over 24 hours shall be used as the additional heat input under the normal and off-normal conditions of storage.

The ambient pressure shall be assumed to be 760mm of Hg coincident with the normal condition temperature, whose bounding value is provided in Table 2.2.2. For sites located substantially above sea level (elevation > 1500 feet ), it will be necessary to perform a site specific evaluation of the peak cladding temperature using the site specific ambient temperature (maximum average annual temperature based on 40 year meteorological data for the site). ISG 11, Revision 3 [2.0.1] temperature limits will continue to apply.

All of the above requirements are consistent with those in the HI-STORM 100 FSAR.

e. 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-STORM FW 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-STORM FW System, under assumed environmental normal temperatures with the maximum decay heat load, result in HI-STORM FW component temperatures at or below the normal condition design temperatures for the HI-STORM FW System defined in Table 2.2.3.

Maintaining fuel rod cladding integrity is also a design consideration. The fuel rod peak cladding temperature (PCT) limits for the long-term storage and short-term operating conditions shall meet the intent of the guidance in ISG-11, Revision 3 [2.0.1]. For moderate burnup fuel the PCT limit for short-term operations is higher than for high burnup fuel [2.0.2].

f. Snow and Ice

The HI-STORM FW System must be capable of withstanding pressure loads due to snow and ice. Section 7.0 of ANSI/ASCE 7-05 [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-STORM FW System range from 50 to 70 pounds per square foot. For conservatism, the snow pressure load (Table 2.2.8) is set to bound the ANSI/ASCE 7-05 recommendation.

## 2.2.2 Loadings Applicable to Off-Normal Conditions

As the HI-STORM FW System is passive, loss of power and instrumentation failures are not defined as off-normal conditions. The off-normal condition design criteria are defined in this subsection.

A discussion of the effects of each off-normal condition and the corrective action for each off-normal condition is provided in Section 12.1. Table 2.2.7 contains a list of all normal and off-normal loadings and their applicable acceptance criteria.

### a. Pressure

The HI-STORM FW System must withstand loads due to off-normal pressure. The off-normal condition for the MPC internal design pressure, defined herein in Table 2.2.1, bounds the cumulative effects of the maximum fill gas volume, off-normal environmental ambient temperatures, the maximum MPC heat load, and an assumed 10% of the fuel rods ruptured with 100% of the fill gas and 30% of the significant radioactive gases (e.g.,  $H^3$ , Kr, and Xe) released as suggested in NUREG-1536.

### b. Environmental Temperatures

The HI-STORM FW 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 to persist for a sufficient duration 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 of temperature used in the definition of the off-normal temperature recognizes the considerable thermal inertia of the HI-STORM FW storage system which essentially flattens the effect of daily temperature variations on the internals of the MPC.

### c. Design Temperatures

In addition to the normal condition design temperatures, which apply to long-term storage and short-term normal operating conditions (e.g., MPC drying operations and onsite transport operations), an off-normal/accident condition temperature pursuant to the provisions of NUREG-1536 and Regulatory Guide 3.61 is also defined. This is the temperature which may exist during a transient event (examples of such an instance is the blockage of the overpack inlet vents or the fire accident). The off-normal/accident condition temperatures of Table 2.2.3 are given to bound

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the maximax (maximum in time and space) value of the thru-thickness average temperature of the structural or non-structural part, as applicable, during the transient event. These enveloping values, therefore, will bound the maximum temperature reached anywhere in the part, excluding skin effects, during or immediately after, a transient event.

The off-normal/accident condition 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 [2.2.4 and 2.2.5]. This ensures that the material will not be subject to significant creep rates during these short duration transient events.

d. 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 confinement boundary is defined by the MPC shell, MPC baseplate, MPC lid, port cover plates, closure ring, and associated welds. 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 weld passes. In addition to multi-pass liquid penetrant examination, the MPC lid-to-shell weld is pressure tested. The vent and drain port cover plates are also subject to proven non-destructive evaluations for leak detection such as liquid penetrant examination. These inspection and testing techniques are performed to verify the integrity of the confinement boundary. Therefore, leakage of one seal is not evaluated for its consequence to the storage system.

e. Partial Blockage of Air Inlets

The loaded HI-STORM FW overpack must withstand the partial blockage of the air inlets. Because the overpack air inlets and outlets are covered by screens and inspected routinely (or alternatively, equipped with temperature monitoring devices), significant blockage of all vents by blowing debris, critters, etc., is very unlikely. Nevertheless, the inherent thermal stability of the HI-STORM FW System shall be demonstrated by assuming all air inlets are partially blocked as an off-normal event.

f. Malfunction of 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 4.6 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 conditions are that the MPC confinement boundary continues to confine the radioactive material, the MPC fuel basket structure maintains the configuration of the contents, the canister can be recovered from the overpack, and the system continues to provide adequate shielding.

A discussion of the effects of each environmental phenomenon and accident condition is provided in Section 12.2. The consequences of each accident or environmental phenomenon are evaluated against the requirements of 10CFR72.106 and 10CFR20. Section 12.2 also provides the corrective action for each event.

#### a. Handling Accident

A handling accident in the Part 72 jurisdiction is precluded by the requirements and provisions specified in this FSAR. The loaded HI-STORM FW components will be lifted in the Part 72 operations jurisdiction in accordance with written and Q.A. validated procedures and shall use lifting devices which comply with ANSI N14.6-1993 [2.2.2] or applicable code. Also, the lifting and handling equipment (typically the cask transporter, which has specific requirements identified in paragraph 1.2.1.5) is required to have a built-in redundancy against uncontrolled lowering of the load. Further, the HI-STORM FW is a vertically deployed system, and the handling evolutions in *short term operations*, as discussed in Chapter 9, do not involve downending of the loaded cask to the horizontal configuration (or upending from the horizontal state) at any time (with the rare handling exception of the transfer cask as described in Subsection 4.5.1). In particular, the loaded MPC shall be lowered into the HI-STORM FW overpack or raised from the overpack using the HI-TRAC VW transfer cask and a MPC lifting system designed in accordance with ANSI N14.6 or applicable code. Therefore, analysis of a handling accident event involving a HI-STORM system component is not required.

b. Non-Mechanistic Tip-Over

The freestanding loaded HI-STORM FW overpack is demonstrated by analysis to remain kinematically stable under all design basis environmental phenomena (tornado, earthquake, etc.) and postulated accident conditions. The cask tip-over is not an outcome of any environmental phenomenon or accident condition and the cask tip-over is considered a *non-mechanistic* event. Nevertheless, the HI-STORM FW overpack and MPC is analyzed for a hypothetical tip-over event, and the structural integrity of a loaded HI-STORM FW System after a tip-over onto a reinforced concrete pad is demonstrated by analysis to show compliance with 10 CFR 72.236(m) with regards to the future transportability of the MPC.

The following requirements and acceptance criteria apply to the HI-STORM FW overpack under the tipover event:

- i. In order to maximize the target stiffness (based on experience with ISFSI pad designs), the ISFSI pad and underlying soil are conservatively modeled using the data in Table 2.2.9.
- ii. The tipover is simulated as a gravity-directed rotation of the cask from rest with its CG above its edge on the pad as the system's initial condition. The tipover begins when the cask is given an infinitesimal outward displacement in the radial plane of its tilted configuration.
- iii. The MPC will remain in the HI-STORM FW overpack after the tipover event and the overpack will not suffer any ovalization which would preclude the removal of the MPC.
- iv. The maximum plastic deformation sustained by the fuel basket panels is limited to the value given in Table 2.2.11.
- v. The HI-STORM FW overpack will not suffer a significant loss of shielding.
- vi. The confinement boundary will not be breached.

c. Fire

The potential of a fire accident near an ISFSI pad is considered to be rendered extremely remote by ensuring that there are no significant combustible materials in the area. The only credible concern is related to a transport vehicle fuel tank fire engulfing the loaded HI-STORM FW overpack or loaded HI-TRAC VW transfer cask while it is being moved to the ISFSI.

The HI-STORM FW System must withstand elevated temperatures due to a fire event. The HI-STORM FW overpack and HI-TRAC VW transfer cask fire accidents for storage are conservatively postulated to be the result of the spillage and ignition of 50 gallons of combustible transporter fuel. The HI-STORM FW overpack and HI-TRAC VW transfer cask surfaces are considered to receive an incident radiation and forced convection heat flux from the fire. Table 2.2.8 provides the fire durations for the HI-STORM FW overpack and HI-TRAC VW transfer cask based on the amount of flammable materials assumed. The temperature of fire is assumed to be 1475° F to accord with the provisions in 10CFR71.73.

The following acceptance criteria apply to the fire accident:

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- i. The peak cladding temperature during and after a fire accident shall not exceed the ISG-11 [2.0.1] permissible limit (see Table 2.2.3).
- ii. The through-thickness average temperature of concrete at any section shall not exceed its short-term limit in Table 2.2.3.
- iii. The steel structure of the overpack shall remain physically stable; i.e., no risk of structural instability such as gross buckling.

d. Partial Blockage of MPC Basket Flow Holes

The HI-STORM FW MPC is designed to prevent reduction of thermosiphon action due to partial blockage of the MPC basket flow holes by fuel cladding failure, fuel debris and crud. The HI-STORM FW 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 of fuel classified as undamaged during storage in the HI-STORM FW. Fuel classified as damaged fuel or fuel debris are placed in damaged fuel containers. The damaged fuel container is equipped with mesh screens which ensure that the damaged fuel and fuel debris will not escape to block the MPC basket flow holes. The MPC is loaded once for long-term storage and, therefore, buildup of crud in the MPC due to numerous loadings is precluded. Using crud quantities for fuel assemblies reported in an Empire State Electric Energy Research Corporation Report [2.2.6] determines a layer of crud of conservative depth that is assumed to partially block the MPC basket flow holes. The crud depth is listed in Table 2.2.8. The flow holes in the bottom of the fuel basket are designed (as can be seen on the licensing drawings) to ensure that this amount of crud does not block the internal helium circulation.

e. Tornado

The HI-STORM FW System must withstand pressures, wind loads, and missiles generated by a tornado. The prescribed design basis tornado and wind loads for the HI-STORM FW System are consistent with NRC Regulatory Guide 1.76 [2.2.7], ANSI 57.9 [2.2.8], and ASCE 7-05 [2.2.3]. Table 2.2.4 provides the wind speeds and pressure drops the HI-STORM FW overpack can withstand while maintaining kinematic stability. The pressure drop is bounded by the accident condition MPC external design pressure.

The kinematic stability of the HI-STORM FW overpack, and continued integrity of the MPC confinement boundary, within the storage overpack or HI-TRAC VW transfer cask, 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 [2.2.9] stipulates that the postulated missiles include at least three objects: a massive high kinetic energy missile that 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 penetrating missile, and a solid sphere for the small rigid missile,

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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.

f. Flood

The HI-STORM FW System must withstand pressure and water forces associated with deep and moving flood waters. Resultant loads on the HI-STORM FW System consist of buoyancy effects, static pressure loads, and velocity pressure due to water velocity. The flood is assumed to deeply submerge the HI-STORM FW System (see Table 2.2.8). The flood water depth is based on the hydrostatic pressure which is bounded by the MPC external pressure stated in Table 2.2.1.

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

The flood water is assumed to be moving. The maximum allowable flood water velocity (Table 2.2.8) is established so that the pressure loading from the water is less than the pressure loading which would cause the HI-STORM FW System to slide or tip over. Site-specific safety reviews by the licensee must confirm that flood parameters at the proposed ISFSI site do not exceed the flood depth or water velocity given in Table 2.2.8.

If the flood water depth exceeds the elevation of the top of the HI-STORM FW overpack inlet vents, then the cooling air flow would be blocked. The flood water may also carry debris which may act to block the air inlets of the overpack. Blockage of the air inlets is addressed in 2.2.3 (l).

The hydrological conditions at most reactor sites are characterized as required by Paragraph 100.10(c) of 10CFR100 and further articulated in Reg. Guide 1.59, "Design Basis Floods for Nuclear Power Plants" and Reg. Guide 1.102, "Flood Protection for Nuclear Power Plants." 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-STORM FW System for kinematic stability, if necessary. An evaluation for tsunamis<sup>†</sup> for certain coastal sites should also be performed to demonstrate that the maximum flood depth in Table 2.2.8 will not be exceeded. The factor of safety against sliding or overturning of the cask under the moving flood waters shall be equal to or greater than the value in Table 2.2.8.

The scenario where the flood water raises high enough to block the inlet ducts (and thus cut-off ventilation) and remains stagnant is the most adverse flood condition (thermally) for the storage system. As discussed in Chapter 1, the HI-STORM FW System inlet vent design makes it resistant to such adverse flood scenarios. The results of this analysis are presented in Chapter 4.

g. Earthquakes

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<sup>†</sup> 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 principal effect of an earthquake on the loaded HI-STORM FW overpack is the movement of the MPC inside the overpack cavity causing impact with the cavity inner wall, and, if the earthquake is sufficiently strong, the potential sliding and tilting of the storage system. The acceptance criteria for the storage system under the site's Design Basis Earthquake (DBE) are as follows:

- i. The loaded overpacks will not impact each other during the DBE event.
- ii. The loaded overpack will not slide off the ISFSI.
- iii. The loaded overpack will not tip over.
- iv. The confinement boundary will not be breached.

To minimize the need for a seismic analysis at each ISFSI site, the approach utilized in Docket No. 72-1014 is adopted for HI-STORM FW, which divides the DBE into two categories, labeled herein as (i) low intensity and (ii) high intensity. A low intensity earthquake is one whose ZPA is low enough to pass the "static equilibrium test". A high intensity earthquake is one that cannot pass the "static equilibrium test". The limiting value of the static friction coefficient,  $\mu$ , has been set at 0.53 for freestanding HI-STORM overpack on a reinforced concrete pad in Docket No. 72-1014. The same limit is observed for HI-STORM FW overpack in this report. The criterion for static equilibrium is derived from elementary statics with the simplifying assumption that the cask and its contents are fixed and emulate a rigid body with six degrees-of-freedom. The earthquake is represented by its ZPA in horizontal (the vector sum of the two horizontal ZPAs for a 3-D earthquake site) and vertical directions. The limits on  $a_H$  and  $a_v$  for HI-STORM FW are readily derived as follows:

- i. Prevention of sliding: Assuming the vertical ZPA to be acting to reduce the weight of the cask, horizontal force equilibrium yields:

$$W \cdot a_H \leq \mu \cdot W \cdot (1 - a_v)$$

$$\text{Or} \quad a_H \leq (1 - a_v) \cdot \mu$$

- ii. Prevention against "edging" of the cask:

Balancing the moment about the cask's pivot point for edging yields:

$$W \cdot a_H \cdot h \leq W \cdot (1 - a_v) \cdot r$$

$$\text{Or} \quad a_H \leq (1 - a_v) \cdot \frac{r}{h}$$

Where:

- r: radius of the footprint of the cask's base
- h: height of the CG of the cask
- $\mu$ : Static friction coefficient between the cask and the ISFSI pad.

The above two inequalities define the limits on  $a_H$  and  $a_v$  for a site if the earthquake is to be considered of “low intensity.” For low intensity earthquake sites, additional analysis to demonstrate integrity of the confinement boundary is not required.

However, if the earthquake’s ZPAs do not satisfy either of the above inequalities, then a dynamic analysis using the methodology specified in Chapter 3 shall be performed as a part of the §72.212 safety evaluation.

h. 100% Fuel Rod Rupture

The HI-STORM FW System must withstand loads due to 100% fuel rod rupture. For conservatism, 100% of the fuel rods are assumed to rupture with 100% of the fill gas and 30% of the significant radioactive gases (e.g.,  $H^3$ , Kr, and Xe) released in accordance with NUREG-1536. All of the fill gas contained in non-fuel hardware, such as burnable poison rod assemblies (BPRAs), is also assumed to be released concomitantly.

i. Confinement Boundary Leakage

None of the postulated environmental phenomenon or accident conditions identified will cause failure of the confinement boundary. Section 7.1 provides the rationale to treat leakage of the radiological contents from the MPC as a non-credible event.

j. External Pressure on the MPC Due to Explosion

The loaded HI-STORM FW overpack must withstand loads due to an explosion. The accident condition MPC external pressure and overpack pressure differential specified in Table 2.2.1 bounds all credible external explosion events. There are no credible internal explosive events since all materials are compatible with the various operating environments, as discussed in Subsection 3.4.1, or appropriate preventive measures are taken to preclude internal explosive events (see Subsection 1.2.1). The MPC is composed of non explosive materials and maintains an inert gas environment. Thus explosion during long term storage is not credible. Likewise, the mandatory use of the protective measures at nuclear plants to prevent fires and explosions and the absence of any need for an explosive material during loading and unloading operations eliminates the scenario of an explosion as a credible event. Furthermore, because the MPC is internally pressurized, any short-term external pressure from explosion or even submergence in flood waters will act to reduce the tensile state of stress in the enclosure vessel. Nevertheless, a design basis external pressure (Table 2.2.1) has been defined as a design basis loading event wherein the internal pressure is non-mechanistically assumed to be absent.

k. Lightning

The HI-STORM FW System must withstand loads due to lightning. The effect of lightning on the HI-STORM FW System is evaluated in Chapter 12.

l. Burial Under Debris and Duct Blockage

Debris may collect on the HI-STORM FW overpack vent screens as a result of floods, wind storms, or mud slides. Siting of the ISFSI pad shall ensure that the storage location is not located over shifting soil. However, if burial under debris is a credible event for an ISFSI, then a thermal analysis to analyze the effect of such an accident condition shall be performed for the site using the analysis methodology presented in Chapter 4. The duration of the burial-under-debris scenario will be based on the ISFSI owner's emergency preparedness program. The following acceptance criteria apply to the burial-under-debris accident event:

- i. The fuel cladding temperature shall not exceed the ISG-11, Revision 3 [2.0.1] temperature limits.
- ii. The internal pressure in the MPC cavity shall not exceed the accident condition design pressure limit in Table 2.2.1.

The burial-under-debris analysis will be performed if applicable, for the site-specific conditions and heat loads.

m. Extreme Environmental Temperature

The HI-STORM FW System must withstand extreme environmental temperatures. The extreme accident level temperature is specified in Table 2.2.2. The extreme accident level temperature is assumed to occur with steady-state insolation. This temperature is assumed to persist for a sufficient duration to allow the system to reach steady-state temperatures. The HI-STORM FW overpack and MPC have a large thermal inertia; therefore, extreme environmental temperature is a 3-day average for the ISFSI site.

All accident events and extreme environmental phenomena loadings that require analysis are listed in Table 2.2.13 along with the applicable acceptance criteria.

The loadings listed in Table 2.2.13 fall into two broad categories; namely, (i) those that primarily affect kinematic stability, and (ii) those that produce significant stresses and strains. 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-STORM FW overpack structure will remain kinematically stable under these loadings. Additionally, for the tornado-borne missile (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 deformations that must be shown to comply with the applicable acceptance criteria. The relevant loading combinations for the fuel basket, the MPC, the HI-TRAC VW transfer cask and the HI-STORM FW overpack are different because of differences in their function. For example, the fuel basket does not experience a pressure loading because it is not a pressure vessel.

## 2.2.4 Applicability of Governing Documents

Section III Subsection NB of the ASME Boiler and Pressure Vessel Code (ASME Code), [2.2.10], is the governing code for the structural design of the MPC. The alternatives to the ASME Code, Section III Subsection NB, applicable to the MPC in Docket Nos. 72-1008 and 72-1014 are also applicable to the MPC in the HI-STORM FW System, as documented in Table 2.2.14.

The stress limits of ASME Section III Subsection NF [2.0.3] are applied to the HI-STORM FW and HI-TRAC VW structural parts where the applicable loading is designated as a code service condition.

The fuel basket, made of Metamic-HT, is subject to the requirements in Chapter 1, Section 1.2.1.4 and is designed to a specific (lateral) deformation limit of its walls under accident conditions of loading (credible and non-mechanistic) (see Table 2.2.11). The basis for the lateral deflection limit in the active fuel region,  $\theta$ , is provided in [2.2.11].

ACI 318 is the reference code for the plain concrete in the HI-STORM FW overpack. ACI 318.1-85(05) is the applicable code utilized to determine the allowable compressive strength of the plain concrete credited in strength analysis.

Each structure, system and component (SSC) of the HI-STORM FW System that is identified as important-to-safety is shown on the licensing drawings.

Tables 1.2.6 and 1.2.7 provide the information on the applicable Codes and Standards for material procurement, design, fabrication and inspection of the components of the HI-STORM FW System. In particular, the ASME Code is relied on to define allowable stresses for structural analyses of Code materials.

## 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:

- i. Level A Service Limits are used to establish allowables for normal condition load combinations.
- ii. Level B Service Limits are used to establish allowables for off-normal conditions.

- iii. Level C Service Limits are not used.
- iv. Level D Service Limits are used to establish allowables for certain accident conditions.

The ASME Code service limits are used in the structural analyses for definition of allowable stresses and allowable stress intensities, as applicable. 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 is required to meet Section III, Class 1, Subsection NB stress intensity limits. Table 2.2.10 lists the stress intensity limits for the Levels A, B, C, and D service limits for Class 1 structures extracted from the ASME Code. Table 2.2.12 lists allowable stress limits for the steel structure of the HI-STORM FW overpack and HI-TRAC VW transfer cask which are analyzed to meet the stress limits of Subsection NF, Class 3 for loadings defined as service levels A, B, and D 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 loads are listed in Tables 2.2.7 and 2.2.13, along with the applicable acceptance criteria.

## 2.2.7 Design Basis Loads

Where appropriate, for each loading type, a bounding value is selected in this FSAR to impute an additional margin for the associated loading events. Such bounding loads are referred to as Design Basis Loads (DBL) in this FSAR. For example, the Design Basis External Pressure on the MPC, set down in Table 2.2.1, is a DBL, as it grossly exceeds any credible external pressure that may be postulated for an ISFSI site.

## 2.2.8 Allowable Limits

The stress intensity limits for the MPC confinement boundary for the design condition and the service conditions are provided in Table 2.2.10. The MPC confinement boundary stress intensity limits are obtained from ASME Code, Section III, Subsection NB. The displacement limit for the MPC fuel basket is expressed as a dimensionless parameter  $\theta$  defined as [2.2.11]

$$\theta = \frac{\delta}{w}$$

where  $\delta$  is defined as the maximum total deflection sustained by the basket panels under the loading event and  $w$  is the nominal inside (width) dimension of the storage cell. The limiting value of  $\theta$  is provided in Table 2.2.11. Finally, the steel structure of the overpack and the HI-

TRAC VW must meet the stress limits of Subsection NF of ASME Code, Section III for the applicable service conditions.

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



Table 2.2.1		
DESIGN PRESSURES		
Pressure Location	Condition	Pressure (psig)
MPC Internal Pressure	Normal	100
	Off-Normal/Short-Term	120
	Accident	200
MPC External Pressure	Normal	(0) Ambient
	Off-Normal/Short-Term	(0) Ambient
	Accident	55
HI-TRAC Water Jacket Internal Pressure	Accident	65
Overpack External Pressure	Normal	(0) Ambient
	Off-Normal/Short-Term	(0) Ambient
	Accident	See Paragraph 3.1.2.1.d

Table 2.2.2		
ENVIRONMENTAL TEMPERATURES		
<b>HI-STORM FW Overpack</b>		
Condition	Temperature (°F)	Comments
Normal Ambient Temperature	80	Bounding annual average from the contiguous United States
Soil Temperature	77	Bounding annual average from the contiguous United States
Off-Normal Ambient Temperature	-40 (min) 100 (max)	Lower bound does not consider insolation.  Upper bound is a 3-day daily average and analysis includes insolation.
Extreme Ambient Temperature	125	3-day daily average and analysis includes insolation
Short-Term Operations	0 (min)	Limit is specified in the technical specifications.
<b>HI-TRAC VW Transfer Cask</b>		
Condition	Temperature (°F)	Comments
Short-Term Operations (Outside)	0 (min.) 90 (max.)	The lower bound limit is specified in the technical specifications. The upper bound limit is a 3-day daily average with insolation and can be increased for a specific site if justified by the appropriate thermal analysis.
Short-Term Operations (Inside a Building)	0 (min.) 110 (max.)	The lower bound limit is specified in the technical specifications. The ambient temperature limit is a 3-day daily average and can be increased for a specific site if justified by the appropriate thermal analysis. Solar insolation is not applicable inside a building.

Table 2.2.3		
DESIGN TEMPERATURES		
HI-STORM FW Component	Normal Condition Design Temperature Limits (°F)	Off-Normal and Accident Condition Temperature Limits <sup>†</sup> (°F)
MPC shell	600	800
MPC basket	752	932
MPC basket shims	752	932
MPC lid	600	800
MPC closure ring	500	800
MPC baseplate	400	800
HI-TRAC VW inner shell	500	700
HI-TRAC VW bottom lid	350	700
HI-TRAC VW top flange	400	650
HI-TRAC VW bottom lid seals	350	N/A
HI-TRAC VW bottom lid bolts	350	800
HI-TRAC VW bottom flange	350	700
HI-TRAC VW radial neutron shield	311	N/A
HI-TRAC VW radial lead gamma shield	350	600
Fuel Cladding	752 (Storage) 752 or 1058 (Short Term Operations) <sup>††</sup>	1058 (Off-Normal and Accident Conditions)
Overpack concrete	300	350
Overpack Lid Top and Bottom Plate	450	700
Remainder of overpack steel structure	350	700

<sup>†</sup> For accident conditions that involve heating of the steel structures and no mechanical loading (such as the blocked air duct accident), the permissible metal temperature of the steel parts is defined by Table 1A of ASME Section II (Part D) for Section III, Class 3 materials as 700°F. For the fire event, the structure is required to remain physically stable (no specific temperature limits apply)

<sup>††</sup> Short term operations include MPC drying and onsite transport. The 1058°F 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.

Table 2.2.4	
CHARACTERISTICS OF REFERENCE TORNADO	
Condition	Value
Rotational wind speed (mph)	290
Translational speed (mph)	70
Maximum wind speed (mph)	360
Pressure drop (psi)	3.0

Table 2.2.5		
TORNADO-GENERATED MISSILES		
Missile Description	Mass (kg)	Velocity (mph)
Automobile	1800	126
Rigid solid steel cylinder (8 in. diameter)	125	126
Solid sphere (1 in. diameter)	0.22	126

Table 2.2.6 LIFTING ANALYSIS CASES					
Loading Case	Item	Location of Threaded Anchor (Material)	Bounding Weight	Dynamic Amplification Factor	Permissible Stress (psi) (Note 1)
HA.	Loaded MPC	Top Lid (stainless steel)	Section 3.2	1.15	Lesser of $0.1 S_u$ or $S_y/3$
HB.	Loaded HI-TRAC Transfer Cask	Top Flange of the Cask (C.S. forging)	Section 3.2	1.15	Lesser of $0.1 S_u$ or $S_y/3$
HC.	Loaded HI-STORM FW Module with Lid	Threaded cylinder embedded and welded to the radial connectors near the top of the cask (carbon steel forging)	Section 3.2	1.15	$S_y/3$
Note 1: The permissible stress applies to the material of the part in which the lift anchor location is tapped. Minimum threaded length of the top shall be used in the analysis. $S_u$ = ultimate strength; $S_y$ = yield strength					

<p style="text-align: center;">Table 2.2.7</p> <p style="text-align: center;">LOADS APPLICABLE TO THE NORMAL AND OFF-NORMAL CONDITIONS OF STORAGE</p>				
Loading Case	Loading	Affected Item and Part	Magnitude of Loading	Acceptance Criterion
NA.	Snow and Ice	Top lid of HI-STORM FW overpack	Table 2.2.8	The stress in the steel structure must meet NF Class 3 limits for linear structures
NB.	Internal Pressure**	MPC Enclosure Vessel	Table 2.2.1	Meet “NB” stress intensity limits
	a. Normal Condition	MPC Enclosure Vessel	Table 2.2.1	Level A condition limit on primary plus secondary stress intensities
	b. Off-Normal Condition	MPC Enclosure Vessel	Table 2.2.1	Level B limits on primary and secondary stress intensities.

\*\* Normal condition internal pressure is bounded by the Design Internal Pressure in Table 2.2.1. Because the top and bottom extremities of the MPC Enclosure Vessel are each at a uniform temperature due to the recirculating helium, thermal stresses are minimal. Therefore, the Design Internal Pressure envelops the case of the Normal Service condition for the MPC. The same remark applies to the Off-Normal Service condition.

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Table 2.2.8		
ADDITIONAL DESIGN INPUT DATA FOR NORMAL, OFF-NORMAL, AND ACCIDENT CONDITIONS		
Item	Condition	Value
Snow Pressure Loading (lb/ft <sup>2</sup> )	Normal	100
Assumed Blockage of MPC Basket Flow Opening by Crud Settling (Depth of Crud, in.)	Accident	1
Cask Environment During the Postulated Fire Event (Deg. F)	Accident	1475
HI-STORM FW Overpack Fire Duration (seconds)	Accident	208
HI-TRAC VW Transfer Cask Fire Duration (minutes)	Accident	4.64
Maximum Submergence Depth due to Flood (ft)	Accident	125
Factor of safety against sliding or overturning from moving flood waters	Accident	1.1



Table 2.2.9

## ISFSI PAD DATA FOR NON-MECHANISTIC TIP-OVER ANALYSIS

Thickness (inch)	36
Concrete Pad Compressive Strength (psi)	6,000
Modulus of elasticity of the subgrade (psi)	28,000

Table 2.2.10			
MPC CONFINEMENT BOUNDARY STRESS INTENSITY LIMITS FOR DIFFERENT LOADING CONDITIONS (ELASTIC ANALYSIS PER NB-3220) <sup>†</sup>			
Stress Category	Design	Level A	Level D <sup>††</sup>
Primary Membrane, $P_m$	$S_m$	$S_m$	AMIN ( $2.4S_m$ , $.7S_u$ )
Local Membrane, $P_L$	$1.5S_m$	$1.5S_m$	150% of $P_m$ Limit
Membrane plus Primary Bending	$1.5S_m$	$1.5S_m$	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 <sup>†††</sup>	$0.6S_m$	$0.6S_m$	$0.42S_u$

<sup>†</sup> Stress combinations including F (peak stress) apply to fatigue evaluations only.

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

<sup>†††</sup> Governed by NB-3227.2 or F-1331.1(d).

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Table 2.2.11	
STRUCTURAL DESIGN CRITERIA FOR THE FUEL BASKET	
PARAMETER	VALUE
Minimum service temperature	-40°F
Maximum total (lateral) deflection in the active fuel region - dimensionless	0.005

Table 2.2.12  
STRESS AND ACCEPTANCE LIMITS FOR DIFFERENT  
LOADING CONDITIONS FOR THE STEEL STRUCTURE OF THE  
HI-STORM FW OVERPACK AND HI-TRAC VW

STRESS CATEGORY	DESIGN + NORMAL	OFF-NORMAL	ACCIDENT <sup>†</sup>
Primary Membrane, $P_m$	S	$1.33 \cdot S$	See footnote
Primary Membrane, $P_m$ , plus Primary Bending, $P_b$	$1.5 \cdot S$	$1.995 \cdot S$	See footnote
Shear Stress (Average)	$0.6 \cdot S$	$0.6 \cdot S$	See footnote

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 Stress

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<sup>†</sup> Under accident conditions, the cask must maintain its physical integrity, the loss of solid shielding (lead, concrete, steel, as applicable) shall be minimal and the MPC must remain recoverable.

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Table 2.2.13 LOADING EVENTS AND ACCEPTANCE CRITERIA APPLICABLE TO ACCIDENT CONDITIONS AND EXTREME ENVIRONMENTAL PHENOMENA				
Loading Case	Loading or Event	Affected Item or Part	Characteristics of Loading	Notes and Acceptance Criterion
AA.	Non-Mechanistic Tip-Over	HI-STORM FW overpack, Fuel Basket and Enclosure Vessel	Impactive load from the slap-down of the loaded overpack	See Paragraph 2.2.3(b)
AB.	Fire	Fuel Cladding, Shielding Concrete, and FW overpack steel structure	Significant radiant heat input over a short time	See Paragraph 2.2.3(c)
AC.	Tornado-Borne Missile	HI-STORM FW overpack	Impactive loading (Table 2.2.5)	See Paragraph 2.2.3(e)
	a. Large Missile	HI-STORM FW overpack	Acting to tip-over the loaded overpack	Use lower bound cask weight, demonstrate kinematic stability
	b. Medium Missile	HI-STORM FW overpack	May damage shielding concrete	Use lower bound cask weight, demonstrate kinematic stability
	c. Small Missile	HI-STORM FW overpack	Penetration	Prevent penetration of the cask and access to the MPC
AD.	Moving Floodwaters	Loaded Storage Module	Acting to tip-over the loaded overpack (Table 2.2.8)	See Paragraph 2.2.3 (f). Use both lower bound and upper bound cask height and weight to demonstrate kinematic stability.
AE.	Design Basis Earthquake	Loaded Storage Module	Acting to destabilize the cask	See Paragraph 2.2.3(g).

<p>Table 2.2.13</p> <p>LOADING EVENTS AND ACCEPTANCE CRITERIA APPLICABLE TO ACCIDENT CONDITIONS AND EXTREME ENVIRONMENTAL PHENOMENA</p>				
AF.	100% Rod Rupture	MPC confinement boundary	Acts to overpressure the MPC and raise the temperature of the fuel cladding	See Paragraph 2.2.3(h). Demonstrate that the equilibrium pressure in the MPC remains below the Accident Condition Design Pressure (Table 2.2.1) and ISG-11 temperature limits are met by the fuel cladding.
AG.	Burial Under Debris	Stored SNF	Blocks convection and retards conduction as means for heat dissipation	See Paragraph 2.2.3(i). Determine the permissible time elapsed under debris so that the pressure in the MPC does not exceed the Accident Condition Design Pressure and the fuel cladding temperature remains below the ISG-11 limit.
AH	Design Basis External Pressure	MPC Enclosure Vessel	An assumed non-mechanistic load from deep submergence in flood water or explosion in the vicinity of the ISFSI	Demonstrate that the MPC Enclosure Vessel will not buckle, i.e., become structurally unstable
AJ.	Internal pressure developed in the HI-TRAC water jacket	HI-TRAC Water Jacket	A non-mechanistic (postulated) event	The water jacket will meet Level D stress limits for "NF" components.

**TABLE 2.2.14**  
**List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)**

MPC Enclosure Vessel	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.	<p>Because the MPC is not an ASME Code stamped vessel, none of the specifications, reports, certificates, or other general requirements specified by NCA are required. In lieu of a Design Specification and Design Report, the HI-STORM FSAR includes the design criteria, service conditions, and load combinations for the design and operation of the MPCs 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.</p> <p>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 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.</p>
MPC Enclosure Vessel	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 basket supports and lift lugs	NB-1130	NB-1132.2(d) requires that the first connecting weld of a non-pressure 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	The lugs that are 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 lug-to-Enclosure Vessel Weld is required to meet the stress limits of Reg. Guide 3.61 in lieu of Subsection NB of the Code.

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**TABLE 2.2.14**  
**List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)**

		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.	
MPC Enclosure Vessel	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 Enclosure Vessel	NB-3100 NF-3100	Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.	These requirements are subsumed by the HI-STORM FW FSAR, serving as the Design Specification, which establishes the service conditions and load combinations for the storage system.
MPC Enclosure Vessel	NB-4120	NB-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, and coating are not, unless explicitly stated by the Code, defined as heat treatment operations.
MPC Enclosure Vessel	NB-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-transfer cask) are guaranteed through fixture-controlled manufacturing. The 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.

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**TABLE 2.2.14**  
**List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)**

MPC Enclosure Vessel	NB-4122	Implies that with the exception of studs, bolts, nuts and heat exchanger tubes, CMTRs must be traceable to a specific piece of material in a component.	MPCs are built in lots. Material traceability on raw materials to a heat number and corresponding CMTR is maintained by Holtec through markings on the raw material. Where material is cut or processed, markings are transferred accordingly to assure traceability. As materials are assembled into the lot of MPCs being manufactured, documentation is maintained to identify the heat numbers of materials being used for that item in the multiple MPCs being manufactured under that lot. A specific item within a specific MPC will have a number of heat numbers identified as possibly being used for the item in that particular MPC of which one or more of those heat numbers (and corresponding CMTRs) will have actually been used. All of the heat numbers identified will comply with the requirements for the particular item.
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.
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 closure ring provides independent redundant closure for vent and drain cover plates. Vent and drain port cover plate welds are helium leakage tested.
MPC Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required.	Only progressive liquid penetrant (PT) examination is permitted. PT examination will include the root and final weld layers and each approx. 3/8" of weld depth.
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested.	The MPC vessel is seal welded in the field following fuel assembly loading. The MPC vessel shall then be pressure tested as defined in Chapter 10. Accessibility for leakage inspections preclude a Code compliant pressure test. All MPC enclosure vessel welds (except closure ring and vent/drain cover plate) are inspected by volumetric examination. MPC shell and shell to baseplate welds are subject to a fabrication helium leak test prior to loading.

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**TABLE 2.2.14**  
**List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)**

			<p>The MPC lid-to-shell weld shall be verified by progressive PT examination. PT must include the root and final layers and each approximately 3/8 inch of weld depth.</p> <p>The inspection results, including relevant findings (indications) shall be made a permanent part of the user's records by video, photographic, or other means which provide an equivalent 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 vent/drain cover plate and the closure ring welds are confirmed by liquid penetrant examination. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME Code Section III, NB-5350.</p>
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.	The HI-STORM FW System is 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.

## 2.3 SAFETY PROTECTION SYSTEMS

### 2.3.1 General

The HI-STORM FW System is engineered to provide for the safe long-term storage of spent nuclear fuel (SNF). The HI-STORM FW will withstand all normal, off-normal, and postulated accident conditions without 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 normal and off-normal operating conditions and its retrievability for further processing or ultimate disposal in accordance with 10 CFR 72.122(l) and ISG-2 [2.3.1].

### 2.3.2 Protection by Multiple Confinement Barriers and Systems

#### 2.3.2.1 Confinement Barriers and Systems

The radioactivity which the HI-STORM FW System must confine originates from the spent fuel assemblies and, to a lesser extent, any radioactive particles from contaminated water in the fuel pool which may remain inside the MPC. This radioactivity is confined by multiple engineered barriers.

Contamination on the outside of the MPC 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 HI-TRAC VW, and the elastomer seal in the HI-TRAC VW bottom lid (see Chapter 9) prevent the fuel pool water from contacting the exterior of the MPC and interior of the HI-TRAC VW while submerged for fuel loading.

The MPC is a seal welded enclosure which provides the confinement boundary. The MPC confinement boundary is defined by the MPC baseplate, MPC shell, MPC lid, closure ring, port cover plates, and associated welds.

The MPC confinement boundary has been designed to withstand any postulated off-normal operations, accident conditions, or external natural phenomena. 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, helium leakage testing of the port cover plate welds, and pressure testing are performed to verify the confinement function. Fabrication inspections and tests are also performed, as discussed in Chapter 10, to verify the integrity of the confinement boundary.

### 2.3.2.2 Cask Cooling

To ensure that an effective passive heat removal capability exists for long term satisfactory performance, several thermal design features are incorporated in the storage system. They are as follows:

- The MPC fuel basket is formed by a honeycomb structure of Metamic-HT plates which allows the unimpeded conduction of heat from the center of the basket to the periphery. The MPC cavity is equipped with the capability to circulate helium internally by natural buoyancy effects and transport heat from the interior region of the canister to the peripheral region (Holtec Patent 5,898,747).
- The MPC confinement boundary ensures that the inert gas (helium) atmosphere inside the MPC is maintained during normal, off-normal, and accident conditions of storage and transfer. 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 ISG-11 limits such that fuel cladding does not experience degradation during the long term storage period.
- The HI-STORM FW is optimally designed, with cooling vents and an MPC to overpack annulus, which maximize air flow by ensuring a turbulent flow regime at maximum heat loads.
- Eight inlet ducts located circumferentially around the bottom of the overpack and the outlet vent which circumscribes the entire lid of HI-STORM FW render the ventilation action insensitive to shifting wind conditions.

### 2.3.3 Protection by Equipment and Instrumentation Selection

#### 2.3.3.1 Equipment

Design criteria for the HI-STORM FW System are described in Section 2.2. The HI-STORM FW System may include use of ancillary or support equipment for ISFSI implementation. Ancillary equipment and structures utilized outside of the reactor facility 10CFR Part 50 structures may be broken down into two broad categories, namely Important-to-Safety (ITS) ancillary equipment and Not Important to Safety (NITS) ancillary equipment. NUREG/CR-6407 provides guidance for the determination of a component's safety classification [1.1.4].

Users may perform the MPC transfer between the HI-TRAC VW transfer cask and the HI-STORM FW overpack in a location of their choice, depending upon site-specific needs and capabilities. For those users choosing to perform the MPC transfer using devices not integral to structures governed by the regulations of 10 CFR Part 50 (e.g., fuel handling or reactor building), a Canister Transfer Facility (CTF) is required. The CTF is typically a concrete lined cavity of a suitable depth to stage the overpack inside it so that the top of the cask is near grade level (Holtec Patent 7,139,358B2). With the overpack staged inside the cavity, the mating device is installed on top and the HI-TRAC VW is mounted on top of the mating device. The MPC

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transfer is carried out by actuating the mating device and moving the MPC vertically to the cylindrical cavity of the recipient cask. The mating device is actuated by removing the bottom lid of the HI-TRAC VW transfer cask (see Figure 1.1.2). The device utilized to lift the HI-TRAC VW transfer cask to place it on the overpack and to vertically transfer the MPC may be of stationary or mobile type, but it must have redundant drop protection features. The cask transporter can be the load handling device at the CTF.

#### 2.3.3.2 Instrumentation

As a consequence of the passive nature of the HI-STORM FW System, instrumentation, which is important to safety, is not necessary. No instrumentation is required or provided for HI-STORM FW storage operations, other than normal security service instruments and dosimeters.

However, in lieu of performing the periodic inspection of the HI-STORM FW overpack vent screens, temperature elements may be installed in the overpack exit vents to continuously monitor the air temperature. If the temperature elements and associated temperature monitoring instrumentation are used, they shall be designated important to safety.

#### 2.3.4 Nuclear Criticality Safety

The criticality safety criteria 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:

- Fuel basket constructed of neutron absorbing material with no potential of detachment.
- Favorable geometry provided by the MPC fuel basket.
- A high B-10 concentration (50% greater than the concentration used in the existing state-of-the art designs certified under 10CFR72) leads to a lower reactivity level under all operating scenarios.

Administrative controls shall be used to ensure that fuel placed in the HI-STORM FW System meets the requirements described in Chapters 2 and 6. 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 is prevented through physical protection means. A security fence surrounded by a physical barrier fence with an appropriate locking and monitoring system is a standard approach to limit access if the ISFSI is located outside the controlled area. 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-STORM FW System.

##### 2.3.5.2 Shielding

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.

The HI-STORM FW is designed to limit dose rates in accordance with 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 doses in excess of the limits given in Table 2.3.1 for normal, off-normal, and accident conditions.

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 loaded overpack are important in consideration of occupational exposure. Conservative evaluations of dose rate have been performed and are described in Chapter 5 based on Reference BWR and PWR fuel (Table 1.0.4).

Consistent with 10 CFR 72, there is no single dose rate limit established for the HI-STORM FW System. Compliance with the regulatory limits on occupational and controlled area doses is performance-based, as demonstrated by dose monitoring performed by each cask user.

Design objective dose rates for the HI-STORM FW overpack surfaces are presented in Table 2.3.2.

Because of the passive nature of the HI-STORM FW System, human activity related to the system after deployment in storage is infrequent and of short duration. Personnel exposures due to operational and maintenance activities are discussed in Chapter 11, wherein measures to reduce occupational dose are also discussed. The estimated occupational doses for personnel provided in Chapter 11 comply with the requirements of 10CFR20. As discussed in Chapter 11, the HI-STORM FW System has been configured to minimize both the site boundary dose in storage and occupational dose during short term operations to the maximum extent possible.

The analyses and discussions presented in Chapters 5, 9, and 11 demonstrate that the HI-STORM FW System is capable of meeting the radiation dose limits set down in Table 2.3.1.

#### 2.3.5.3 Radiological Alarm System

The HI-STORM FW does not require a radiological alarm system. There are no credible events that could result in release of radioactive materials from the system and direct radiation exposure from the ISFSI is monitored using the plant's existing dose monitoring system.

#### 2.3.6 Fire and Explosion Protection

There are no combustible or explosive materials associated with the HI-STORM FW System. Combustible materials will not be stored within an ISFSI. However, for conservatism, a hypothetical fire accident has been analyzed as a bounding condition for HI-STORM FW System. The evaluation of the HI-STORM FW System fire accident is discussed in Chapter 12.

Explosive material will not be stored within an ISFSI. Small overpressures may result from accidents involving explosive materials which are stored or transported in the vicinity of the site. Explosion is an accident loading condition considered in Chapter 12.

Table 2.3.1 RADIOLOGICAL SITE BOUNDARY REQUIREMENTS	
MINIMUM DISTANCE TO BOUNDARY OF CONTROLLED AREA (m)	100
NORMAL AND OFF-NORMAL CONDITIONS:	
-Whole Body (mrem/yr)	25
-Thyroid (mrem/yr)	75
-Any Other Critical Organ (mrem/yr)	25
DESIGN BASIS ACCIDENT:	
-TEDE (rem)	5
-DDE + CDE to any individual organ or tissue (other than lens of the eye) (rem)	50
-Lens dose equivalent (rem)	15
-Shallow dose equivalent to skin or any extremity (rem)	50



Table 2.3.2 – Design Objective Dose Rates for HI-STORM FW Overpack Surfaces	
Area of Interest	Dose Rate (mrem/hr)
Radial Surface Excluding Vents	300
Inlet and Outlet Vents	300
Top of the Lid (Horizontal Surface at approximate center)	30

## 2.4 DECOMMISSIONING CONSIDERATIONS

Efficient decommissioning of the ISFSI is a paramount objective of the HI-STORM FW System. The HI-STORM FW System is ideally configured to facilitate rapid, safe, and economical decommissioning of the storage site. As discussed below, Holtec International has taken appropriate steps to ensure that the necessary equipment designs and certifications shall be available to the user of the HI-STORM FW System to expeditiously decommission the ISFSI at the end of the storage facility's required service life.

Towards that end, the MPC has been designed with the objective to transport it in a HI-STAR 190 transportation cask (Figure 2.4.1). Since the loaded MPC is a self-contained "Waste Package", no further handling of the SNF stored in the MPC is required prior to transport to a licensed centralized storage facility or repository.

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 do not need to be removed. However, to ensure a practical, feasible method to defuel the 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) severing of the MPC closure ring to provide access to the MPC vent and drain. The circumferential welds of the MPC closure lid can be removed by semiautomatic or remotely actuated means, providing access to the SNF.

Likewise, the overpack consists of steel and concrete rendering it suitable for permanent burial. Alternatively, the MPC can be removed from the overpack, and the latter reused for storage of other MPCs. In either case, the overpack would be expected to have no interior or exterior radioactive surface contamination. Any neutron activation of the steel and concrete 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 SNF needs to be removed from the MPC before it is placed into the MGDS, the MPC interior metal surfaces can be decontaminated using existing mechanical or chemical methods to allow for its disposal. This will be facilitated by the 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-STORM FW System may pose is slight activation of the HI-STORM FW materials caused by irradiation over the storage period.

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Due to the design of the HI-STORM FW 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 overpack is removed.

The long-lived radionuclides produced by the irradiation of the HI-STORM FW System components are listed in Table 2.4.1. The activation of the HI-STORM FW components shall be limited to a cumulative activity of 10 Ci per cubic meter before decommissioning and disposal of the activated item can be carried out.

In any case, the HI-STORM FW System would not impose any additional decommissioning requirements on the licensee of the ISFSI facility per 10CFR72.30, since the HI-STORM FW System could eventually be shipped from the site.

Table 2.4.1

**PRINCIPAL LONG-LIVED ISOTOPES PRODUCED DURING IRRADIATION OF THE  
HI-STORM FW COMPONENTS**

<b>Nuclide</b>	<b>MPC Stainless Steel</b>	<b>HI-STORM Steel</b>	<b>HI-STORM Concrete</b>
<sup>54</sup> Mn	X	X	X
<sup>55</sup> Fe	X	X	X
<sup>59</sup> Ni	X	-	-
<sup>60</sup> Co	X	-	-
<sup>63</sup> Ni	X	-	-
<sup>39</sup> Ar	-	-	X
<sup>41</sup> Ca	-	-	X

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**Withheld in Accordance with 10 CFR 2.390**

**Figure 2.4.1: HI-STAR 190 Transportation Overpack and MPC Shown in  
Exploded, Cut-Away View**

## 2.5 REGULATORY COMPLIANCE

Chapter 2 provides the principal design criteria and applicable loading related to HI-STORM FW structures, systems, and components designated as 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 detailed evaluation of the design criteria and an assessment of compliance with those criteria are provided in Chapters 3 through 12.

## 2.6 REFERENCES

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- [2.2.11] Holtec Proprietary Position Paper DS-331, "Structural Acceptance Criteria for the Metamic-HT Fuel Basket", (USNRC Docket No. 71-9325).
- [2.3.1] ISG-2, "Fuel Retrievability", Revision 0, USNRC, Washington DC



## CHAPTER 3: STRUCTURAL EVALUATION<sup>†</sup>

### 3.0 OVERVIEW

In this chapter, the structural components of the HI-STORM FW system subject to certification by the USNRC are identified and described. The objective of the structural analyses is to ensure that the integrity of the HI-STORM FW system is maintained under all credible loadings under normal, off-normal and extreme environmental conditions as well all credible accident events. The results of the structural analyses, summarized in this FSAR, support the conclusion that the confinement, criticality control, radiation shielding, and retrievability criteria set forth under 10CFR72.236(l), 10CFR72.124(a), 10CFR72.104, 10CFR72.106, and 10CFR72.122(l) shall be met by the storage system. In particular, the design basis information contained in the previous two chapters and in this chapter provides the necessary data to permit all needed structural evaluations for demonstrating compliance with the requirements of 10CFR72.236(a), (b), (d) (e), (f), (g), and (l). To facilitate regulatory review, the assumptions and conservatism inherent in the analyses are identified along with a concise description of the analytical methods, models, and acceptance criteria. A summary of the system's ability to maintain its structural integrity under other slow acting (degenerative) or precipitous (sudden) effects that may contribute to structural failure, such as, corrosion, fatigue, buckling, and non-ductile fracture is also provided. The information presented herein is intended to comply with the guidelines of NUREG-1536 and ISG-21 pertaining to use of finite element codes.

In particular, every Computational Modeling Software (CMS) deployed to perform the structural analyses is identified and its implementation appropriately justified as suggested in ISG-21. The information on benchmarking and validation of each Computational Modeling Software is also provided (in Subsection 3.6.2).

Where appropriate, the structural analyses have been performed using classical strength materials solution. Such calculations are presented in this FSAR in transparent detail.

Furthermore, the input data and analyses using Computational Modeling Software (CMS) are described in sufficient detail to enable an independent evaluation of safety conclusions reached in this chapter.

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<sup>†</sup> 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. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the Glossary.

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 unquantified 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.

## 3.1 STRUCTURAL DESIGN

### 3.1.1 Discussion

The HI-STORM FW system consists of the Multi-Purpose Canister (MPC) and the storage overpack (Figure 1.1.1). The components subject to certification on this docket consist of the HI-STORM FW system components and the HI-TRAC VW transfer cask (please see Table 1.0.1). A complete description of the design details of these three components are provided in Section 1.2. This section discusses the structural aspects of the MPC, the storage overpack, and the HI-TRAC VW transfer cask. Detailed licensing drawings for each component are provided in Section 1.5.

#### (i) The Multi-Purpose Canister (MPC)

The design of the MPC seeks to attain three objectives that are central to its functional adequacy:

- **Ability to Dissipate Heat:** The thermal energy produced by the stored spent fuel must be transported to the outside surface of the MPC to maintain the fuel cladding and fuel basket metal walls below the regulatory temperature limits.
- **Ability to Withstand Large Impact Loads:** The MPC, with its payload of nuclear fuel, must withstand the large impact loads associated with the non-mechanistic tipover event.
- **Restraint of Free End Expansion:** The MPC structure is designed so that membrane and bending (primary) stresses produced by constrained thermal expansion of the fuel basket do not arise.

As stated in Chapter 1, the MPC Enclosure Vessel is a confinement vessel designed to meet the stress limits in ASME Code, Section III, Subsection NB. The enveloping canister shell, baseplate, and the lid system form a complete Confinement Boundary for the stored fuel that is referred to as the "Enclosure Vessel". Within this cylindrical shell confinement vessel is an egg-crate assemblage of Metamic-HT plates that form prismatic cells with square cross sectional openings for fuel storage, referred to as the fuel basket. All multi-purpose canisters designed for deployment in the HI-STORM FW have identical external diameters. The essential difference between the different MPCs lies in the fuel baskets, each of which is designed to house different types of fuel assemblies. All fuel basket designs are configured to maximize structural integrity through extensive inter-cell connectivity. Although all fuel basket designs are structurally similar, analyses for each of the MPC types is carried out separately to ensure structural compliance.

The design criteria of components in the HI-STORM FW system important to safety are defined in Chapter 2.

The principal structural functions of the MPC in storage mode are:

- i. To position the fuel in a subcritical configuration, and
- ii. To provide a leak tight Confinement Boundary.

The key structural functions of the overpack during storage are:

- i. To serve as a missile barrier for the MPC,
- ii. To provide flow paths for natural convection,
- iii. To provide a kinematically stable SNF storage configuration,
- iv. To provide fixed and reliable radiation shielding, and
- v. To allow safe translocation of the overpack with a loaded MPC inside.

Some structural features of the MPCs that 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 mechanical sealing arrangement except welding. The absence of any gasketed or flanged joints makes the MPC structure immune from 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 the closure ring. The MPC lid can be either a single thick circular plate continuously welded to the MPC shell along its circumference or a two-piece lid, dual lids welded around their common periphery. When using a two piece lid only the top portion of the lid is considered as part of the closure system, the bottom portion is only for shielding purposes. The MPC closure system is shown in the licensing drawings in Section 1.5. The MPC lid is equipped with vent and drain ports, which are used both for evacuating moisture and air from the MPC following fuel loading and subsequent backfilling with an inert gas (helium) at a specified mass. 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 lid and shell. The two closure members are interconnected by welding around the inner diameter of the ring. Lift points for the MPC are provided on the MPC lid.
- The MPC fuel baskets consist of an array of interconnecting plates. The number of storage cells formed by this interconnection process varies depending on the type of fuel being stored. Basket configurations designed for both PWR and BWR fuel are explained in detail in Section 1.2. All baskets are designed to fit into the same MPC shell.

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- The MPC basket is separated from its lateral supports (basket shims) by a small, calibrated gap designed to prevent thermal stressing associated with the thermal expansion mismatches between the fuel basket and the basket support structure. The gap is designed to ensure that the basket remains unconstrained when subjected to the thermal heat generated by the spent nuclear fuel.

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 non-mechanistic tipover event. This requirement is satisfied if the MPC fuel basket plates undergo a minimal deflection (see Table 2.2.11). The fuel basket strains are shown in Subsection 3.4.4.1.4 to remain essentially elastic, and, therefore, there is no impairment in the recoverability or retrievability of the fuel and the subcriticality of the stored fuel is unchallenged.

The MPC Confinement Boundary contains no valves or other pressure relief devices. In addition, the analyses presented in Subsections 3.4.3, 3.4.4.1.5, and 3.4.4.1.6 show that the MPC Enclosure Vessel meets the stress intensity criteria of the ASME Code, Section III, Subsection NB for all service conditions. Therefore, the demonstration that the MPC Enclosure Vessel meets Subsection NB stress limits ensures that there will be no discernible release of radioactive materials from the MPC.

#### (ii) Storage Overpack

The HI-STORM FW storage overpack is a steel cylindrical structure consisting of inner and outer low carbon steel shells, a lid, and a baseplate. Between the two shells is a thick cylinder of unreinforced (plain) concrete. Plain concrete is also installed in the lid to minimize skyshine. The storage overpack serves as a missile and radiation barrier, provides flow paths for natural convection, provides kinematic stability to the system, and acts as a shock absorber for the MPC in the event of a postulated tipover accident. The storage overpack is not a pressure vessel since it contains cooling vents. The structural steel weldment of the HI-STORM FW overpack is designed to meet the stress limits of the ASME Code, Section III, Subsection NF, Class 3 for normal and off-normal loading conditions and Regulatory Guide 3.61 for handling conditions.

As discussed in Chapters 1 and 2, the principal shielding material utilized in the HI-STORM FW overpack is plain concrete. The plain concrete in the HI-STORM FW serves a structural function only to the extent that it may participate in supporting direct compressive or punching loads. The allowable compression/bearing resistance is defined and quantified in ACI -318-05 [3.3.5]. Strength analyses of the HI-STORM FW overpack and its confined concrete have been carried out in Subsections 3.4.4.1.3 and 3.4.4.1.4 to show that the concrete is able to perform its radiation protection function and that retrievability of the MPC subsequent to any postulated accident condition of storage or handling is maintained.

### (iii) Transfer Cask

The HI-TRAC VW transfer cask is the third component type subject to certification. Strictly speaking, the transfer cask is an ancillary equipment which serves to enable the *short term operations* to be carried out safely and ALARA. Specifically, the transfer cask provides a missile and radiation barrier during transport of the MPC from the fuel pool to the HI-STORM FW overpack. Because of its critical role in insuring a safe dry storage implementation, the transfer cask is subject to certification under 10CFR 72 even though it is not a device for storing spent fuel.

The HI-TRAC VW body is a double-walled steel cylinder that constitutes its structural system. Contained between the two steel shells is an intermediate lead cylinder. Integral to the exterior of the HI-TRAC VW body outer shell is a water jacket that acts as a radiation barrier. The HI-TRAC VW is not a pressure vessel since it contains penetrations and openings. The structural steel components of the HI-TRAC VW are subject to the stress limits of the ASME Code, Section III, Subsection NF, Class 3 for normal and off-normal loading conditions.

Since the HI-TRAC VW may serve as an MPC carrier, its lifting attachments are designed to meet the design safety factor requirements of NUREG-0612 [3.1.1] and Regulatory Guide 3.61 [1.0.2] for single-failure-proof lifting equipment.

### 3.1.2 Design Criteria and Applicable Loads

Principal design criteria for normal, off-normal, and accident/environmental events are discussed in Section 2.2. In this section, the loads, load combinations, and the structural performance of the HI-STORM FW system under the required loading events 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-STORM FW system possesses sufficient structural capability to withstand normal and off-normal loads and the worst case loads under natural phenomenon or accident events. Withstanding such loadings implies that the HI-STORM FW system will successfully preclude the following:

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

The above design objectives for the HI-STORM FW system can be particularized for individual components as follows:

- The objectives of the structural analysis of the MPC are to demonstrate that:
  - i. Confinement of radioactive material is maintained under normal, off-normal, accident conditions, and natural phenomenon events.
  - ii. The MPC basket does not deform under credible loading conditions such that the subcriticality or retrievability of the SNF is jeopardized.
- The objectives of the structural analysis of the storage overpack are to demonstrate that:
  - i. Large energetic missiles such as tornado-generated missiles do not compromise the integrity of the MPC Confinement Boundary.
  - ii. The radiation shielding remains properly positioned in the case of any normal, off-normal, or natural phenomenon or accident event.
  - iii. The flow path for the cooling airflow shall remain available under normal and off-normal conditions of storage and after a natural phenomenon or accident event.
  - iv. The loads arising from normal, off-normal, and accident level conditions exerted on the contained MPC do not violate the structural design criteria of the MPC.
  - v. No geometry changes occur under any normal, off-normal, and accident level conditions of storage that preclude ready retrievability of the contained MPC.
  - vi. A freestanding storage overpack loaded with a MPC can safely withstand a non-mechanistic tip-over event.
  - vii. The inter-cask transfer of a loaded MPC can be carried out without exceeding the structural capacity of the HI-STORM FW overpack, provided all required auxiliary equipment and components specific to an ISFSI site comply with their design criteria set forth in this FSAR and the handling operations are in full compliance with operational limits and controls prescribed in this FSAR.

- The objective of the structural analysis of the HI-TRAC VW transfer cask is to demonstrate that:
  - i. Tornado generated missiles do not compromise the integrity of the MPC Confinement Boundary while the MPC is contained within HI-TRAC VW.
  - ii. No geometry changes occur under any postulated handling or storage conditions that may preclude ready retrievability of the contained MPC.
  - iii. The structural components perform their intended function during lifting and handling with the loaded MPC.
  - iv. The radiation shielding remains properly positioned under all applicable handling service conditions for HI-TRAC VW.

The above design objectives are deemed to be satisfied for the MPC, the overpack, and the HI-TRAC VW, if stresses (or stress intensities or strains, as applicable) calculated by the appropriate structural analyses are less than the allowables defined in Subsection 3.1.2.3, and if the diametral change in the storage overpack (or HI-TRAC VW), if any, after any event of structural consequence to the overpack (or transfer cask), does not preclude ready retrievability of the contained MPC.

Stresses arise in the components of the HI-STORM FW system due to various loads that originate under normal, off-normal, or accident conditions. These individual loads are combined to form load combinations. Stresses, strains, displacements, and stress intensities, as applicable, resulting from the load combinations are compared to their respective allowable limits. The following subsections present loads, load combinations, and the allowable limits germane to them for use in the structural analyses of the MPC, the overpack, and the HI-TRAC VW transfer cask.

#### **3.1.2.1 Applicable Loadings**

The individual loads applicable to the HI-STORM FW system and the HI-TRAC VW cask are defined in Section 2.2 of this FSAR. Load combinations are developed by assembling the individual loads that may act concurrently, and possibly, synergistically. In this subsection, the individual loads are further clarified as appropriate and the required load combinations are identified. Table 3.1.1 contains the governing load cases and the affected components. Loadings are applied to the mathematical models of the MPCs, the overpack, and the HI-TRAC VW. Results of the analyses carried out under bounding load combinations are compared with their respective allowable limits. The analysis results from the bounding load combinations are also evaluated to ensure satisfaction of the functional performance criteria discussed in the foregoing.

The individual loads that address each design criterion applicable to the structural design of the HI-STORM FW system are cataloged in Tables 2.2.6, 2.2.7, and 2.2.13 for the handling, normal, off-normal, and accident (Design Basis Loads) conditions, respectively. The magnitude of loadings

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associated with accident condition and natural phenomena-induced events, in general, do not have a regulatory 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, representative magnitudes of such loadings are drawn from regulatory and industry documents (such as for tornado missiles and wind from Reg. Guide 1.76). In the following, the essential characteristics of both credible and non-credible loadings analyzed in this FSAR are explained.

a. Tip-Over

The freestanding HI-STORM FW storage overpack, containing a loaded MPC, must not tip over as a result of postulated natural phenomenon events, 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 is analyzed (Subsection 2.2.3) in this chapter. For MPC transfers that will occur outside of a Part 50 controlled structure, the potential of the HI-STORM FW overpack tipping over during the lowering (or raising) of the loaded MPC from (or into) the mounted HI-TRAC VW cask is ruled out because of the safeguards and devices mandated by this FSAR for such operations (Subsection 2.3.3). The physical and procedural barriers imposed during MPC handling operations, as described in this FSAR, prevent overturning of the HI-STORM/HI-TRAC assemblage with an extremely high level of certainty. Among the physical barriers to prevent the overturning of the HI-STORM/HI-TRAC stack during MPC transfer is the use of the Canister Transfer Facility illustrated in Figure 1.1.2 which secures the HI-STORM FW inside an engineered pit.

b. Handling Accident

The handling of all heavy loads that are within Part 72 jurisdiction must be carried out using high integrity handling equipment, and single failure-proof lifting devices to render an uncontrolled lowering of the load non-credible (please see Subsection 2.2.3).

c. Flood

Flood at an ISFSI is designated as an extreme environmental event and is described in Subsection 2.2.3 (f).

The postulated flood event has two discrete potential structural consequences; namely,

- i. stability of the HI-STORM FW system due to flood water velocity, and
- ii. structural effects of hydrostatic pressure and water velocity induced lateral pressure.

The maximum hydrostatic pressure on the cask in a flood where the water level is conservatively set per Table 2.2.8 is calculated as follows:

Using  $p$  = the maximum hydrostatic pressure on the system (psi),  
 $\gamma$  = weight density of water = 62.4 lb/ft<sup>3</sup>,  
 $h$  = the height of the water level = 125 ft;

The maximum hydrostatic pressure is

$$p = \gamma h = (62.4 \text{ lb/ft}^3)(125 \text{ ft})(1 \text{ ft}^2/144 \text{ in}^2) = 54.2 \text{ psi}$$

It is noted that the accident condition design external pressure for the MPC (Table 2.2.1) bounds the maximum hydrostatic pressure exerted by the flood.

The maximum acceptable water velocity for a moving flood water scenario is computed using the procedure in Subsection 3.4.4.1.1.

d. Explosion

Explosion, by definition, is a transient event. Explosive materials (except for the short duration when a limited quantity of motive fuel for placing the loaded MPC on the ISFSI pad is present in the tow vehicle or transporter) are prohibited in the controlled area by specific stipulation in the HI-STORM FW Technical Specification. However, pressure waves emanating from explosions in areas outside the ISFSI are credible.

Pressure waves from an explosive blast in a property near the ISFSI site result in an impulsive aerodynamic loading on the stored HI-STORM FW overpacks. Depending on the rapidity of the pressure build-up, the inside and outside pressures on the HI-STORM FW METCON™ shell may not equalize, leading to a net lateral loading on the upright overpack as the pressure wave traverses the overpack. The magnitude of the dynamic pressure wave is conservatively set to a value below the magnitude of the pressure differential that would cause a tip-over of the cask if the pulse duration were set infinite.

The allowable pressure from explosion,  $p_e$ , can be computed from static equilibrium to prevent sliding or tipping of the cask. A simplified inequality to ensure that the cask will not slide is given by

$$p_e D L \leq \mu W$$

where:

- D: diameter of the cask
- L: height of the cask above the ISFSI pad
- $\mu$ : limiting value of the interface friction coefficient
- W: weight of the cask (lower bound weight, assuming that the MPC has only one fuel assembly)

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$$p_e \leq \frac{\mu W}{DL} \quad (A)$$

The inequality for protection against tipping is obtained by moment equilibrium.

$$p_e D \frac{L^2}{2} \leq \frac{W D}{2}$$

$$\text{or} \quad p_e \leq \frac{W}{L^2} \quad (B)$$

The allowable value of  $p_e$  must be lesser of the two values given by inequalities (A) and (B) above.

In contrast to the overpack, the MPC is a closed pressure vessel. Because of the enveloping overpack around it, the explosive pressure wave would manifest as an external pressure on the external surface of the MPC.

The maximum overpressure on the MPC resulting from an explosion is limited by the HI-STORM FW Technical Specification to be equal to or less than the accident condition design external pressure specified in Table 2.2.1.

e. Tornado

The tornado loading is described in Subsection 2.2.3 (e). The three components of a tornado load are:

1. pressure changes,
2. wind loads, and
3. tornado-generated missiles.

Reference values of wind speeds and tornado-induced pressure drop are specified in Table 2.2.4. Tornado missiles are listed in Table 2.2.5. A central functional objective of a storage overpack is to maintain the integrity of the “Confinement Boundary”, namely, the multi-purpose canister stored inside it. This operational imperative requires that the mechanical loadings associated with a tornado at the ISFSI do not jeopardize the physical integrity of the loaded MPC. 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 the pressure drop
- Loadings applied on the MPC transmitted to the inside of the overpack through its openings or as a secondary effect of loading on the enveloping overpack structure.
- Excessive storage overpack permanent deformation that may prevent ready retrievability of the MPC.
- Excessive storage overpack permanent deformation that may significantly reduce the shielding effectiveness of the storage overpack.

Analyses must be performed to ensure that, due to the tornado-induced loadings:

- The overpack does not deform plastically such that the retrievability of the stored MPC is threatened.
- The MPC Confinement Boundary is not breached.
- The MPC fuel basket does not deform beyond the permitted limit (Table 2.2.11) to preserve its subcriticality margins (requires evaluation if the overpack tips over).

f. Earthquake

The earthquake loading and the associated acceptance criteria are presented in Subsection 2.2.3(g).

The Design Basis Earthquake for an ISFSI site shall be obtained *on the top surface of the pad* using an appropriate soil-structure interaction Code such as SHAKE2000 [3.1.7]. The seismic analysis methodology is provided in Subsection 3.4.4.1.2.

g. Lightning

The HI-STORM FW overpack contains over 50,000 lb of highly conductive carbon steel with over 700 square feet of external surface area. It is known from experience that such a large surface area and metal mass is adequate to dissipate any lightning that may strike the HI-STORM FW system. There are no combustible materials on the HI-STORM FW surface. Therefore, a postulated lightning strike event will not impair the structural performance of components of the HI-STORM FW system that are important-to-safety.

#### h. Fire

The fire event applicable to an ISFSI is described in Subsection 2.2.3(c) wherein the acceptance criteria are also presented.

#### i. 100% Fuel Rod Rupture

The sole effect of the postulated 100% fuel rod rupture is to increase the internal pressure in the MPC. Calculations in Chapter 4 show that the accident internal pressure limit set in Chapter 2 bounds the pressure from 100% fuel rod rupture. Therefore, 100% rod rupture does not define a new controlling loading event.

### 3.1.2.2 Design Basis Loads and Load Combinations

As discussed in Subsection 2.2.7, the number of discrete loadings 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  $E^*$ . These bounding loads are referred to as “Design Basis Loads”.

The Design Basis Loads are analyzed in combination with other permanent loads, i.e., loads that are present at all times. The permanent loads consist of:

- The dead load of weight of each component.
- Internal pressure in the MPC.

For conservatism, the upper or lower bound of the dead load,  $D$ , of a component is used for a DBL to maximize the response. Thus, the lower bound value of  $D$  is used in the stability of the HI-STORM FW system under flood. Likewise, the value of internal pressure in the MPC is represented by the Design Pressure (Table 2.2.1), which envelops the actual internal pressure under each service condition.

As noted previously, certain loads, namely earthquake  $E$ , flowing water under flood condition  $F$ , force from an explosion pressure pulse  $F^*$ , and tornado missile  $M$ , act to destabilize a cask. Additionally, these loads act on the overpack and produce essentially localized stresses at the HI-STORM FW system to ISFSI interface. Table 3.1.1 provides the load combinations that are relevant to the stability analyses of freestanding casks.

The major constituents in the HI-STORM FW system are: (i) the fuel basket, (ii) the Enclosure Vessel, (iii) the HI-STORM FW overpack, and (iv) the HI-TRAC VW transfer cask. The fuel basket and the Enclosure Vessel (EV) together constitute the multi-purpose canister. A complete account of analyses and results for all applicable loadings for all four constituent parts is provided in Section 3.4 as suggested in Regulatory Guide 3.61.

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In the following, the loadings listed as applicable for each situational condition are addressed in meaningful load combinations for the fuel basket, Enclosure Vessel, and the overpack. Each component is considered separately.

a. Fuel Basket

Table 3.1.1 summarizes the loading cases (derived from Tables 2.2.6, 2.2.7, and 2.2.13) that are germane to demonstrating compliance of the loaded fuel baskets inside the MPC Enclosure Vessel.

The fuel basket is not a pressure vessel; therefore, the pressure loadings are not meaningful loads for the basket. Further, the basket is physically disconnected 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-Enclosure Vessel (EV) gap to ensure absence of interference due to differential thermal expansion is addressed in Chapter 4.

The normal handling of the MPC within the HI-STORM FW system or the HI-TRAC VW transfer cask does not produce any significant stresses in the fuel basket because the operating procedures preclude horizontal handling.

b. Enclosure Vessel

Table 3.1.1 summarizes all load cases that are applicable to structural analysis of the Enclosure Vessel to ensure integrity of the Confinement Boundary.

The Enclosure Vessel is a pressure retaining device 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 per 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 (Level A service condition in the Code).

Normal handling of the Enclosure Vessel is considered in Section 2.2; the handling loads are independent of whether the Enclosure Vessel is within the storage overpack or HI-TRAC VW cask.

c. Storage Overpack

Table 3.1.1 identifies the load cases to be considered for the overpack. The following acceptance criteria apply:

i. Normal Conditions

- The dead load of the HI-TRAC VW with the heaviest loaded MPC (dry) on top of the HI-STORM FW overpack must be shown to be able to be supported by the metal-concrete (METCON™) structure consisting of the two concentric steel shells and the radial ribs.
- The stress field in the steel structure of the overpack must meet Level A (Subsection NF) limits.

ii. Accident Conditions

- Maximum flood water velocity for the overpack with a near empty MPC (only one SNF stored) shall not cause sliding or tip-over of the cask.
- Tornado missile plus wind on an overpack (with an empty MPC) (see Table 2.2.4) must not lead to violation of the acceptance criteria in 3.1.2.1(e).
- Large or medium penetrant missiles (see Table 2.2.5) must not be able to access the MPC. The small missile must be shown not to penetrate the MPC pressure vessel boundary since, in principle, it can enter the overpack cavity through the (curvilinear) vent inlet vent passages.
- Under seismic conditions, a freestanding HI-STORM FW overpack must be demonstrated to not tip over under the DBE events. The maximum sliding of the overpack must demonstrate that casks will not impact each other.
- Under a non-mechanistic tip-over of a fully loaded, freestanding HI-STORM FW overpack, the overpack lid must not dislodge.
- Accident condition induced gross general deformations of the storage overpack must be limited to values that do not prevent ready retrievability of the MPC.

d. HI-TRAC VW Transfer Cask

Table 3.1.1 culled from Tables 2.2.6, 2.2.7 and 2.2.13 identifies load cases applicable to the HI-TRAC VW transfer cask.

The HI-TRAC VW transfer cask must provide radiation protection, must act as a handling cask when carrying a loaded MPC, and in the event of a postulated accident must not suffer permanent deformation to the extent that ready retrievability of the MPC is compromised.

### 3.1.2.3 Allowables

The important-to-safety (ITS) components of the HI-STORM FW system are identified on the drawings in Section 1.5. Allowable stresses, as appropriate, are tabulated for these components for all service conditions.

In Section 2.2, the applicable service level from the ASME Code for determination of allowables is listed. Tables 2.2.6, 2.2.7 and 2.2.13 (condensed in Table 3.1.1) provide a tabulation of loadings for normal, off-normal, and accident conditions and the applicable acceptance criteria.

Relationships for allowable stresses and stress intensities for NB and NF components are provided in Tables 2.2.10 and 2.2.12, respectively. Tables 3.1.2 through 3.1.8 contain numerical values of the allowable stresses/stress intensities for all MPC, overpack, and HI-TRAC VW load bearing Code materials as a function of temperature. The tabulated values for the allowable stresses/stress intensities are used in Subsections 3.4.3 and 3.4.4, as applicable, to compute factors of safety for the ITS components of the HI-STORM FW system for various loadings.

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 that 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 stress analysis of the HI-STORM FW system extracted from the ASME Code (see Figure NB-3222-1, for example) are listed in Table 3.1.10.

### 3.1.2.4 Brittle Fracture

Section 8.4.3 discusses the low temperature ductility of the HI-STORM FW system materials. Table 3.1.9 provides a summary of impact testing requirements to insure prevention of brittle fracture.

### 3.1.2.5 Fatigue

Fatigue is a consequence of a cyclic state of stress applied on a metal part. Failure from fatigue occurs if the combination of amplitude of the cyclic stress,  $\sigma_a$ , and the number of cycles,  $n_f$ , reaches a threshold value at which failure occurs. ASME Code, Section III, Subsection NCA provides the  $\sigma_a$ - $n_f$  curves for a number of material types. At  $n_f = 10^6$ , the required  $\sigma_a$  is referred to as the “Endurance Limit”. The Endurance Limit for stainless steel (the material used in the MPC) according to the ASME Code, Section III, Div. 1, Appendices, Table I.9.2, is approximately 28 ksi.

The causative factors for fatigue expenditure in a non-active system (i.e., no moving parts) such as the HI-STORM FW system may be:

- i. rapid temperature changes
- ii. significant pressure changes

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The HI-STORM FW system is exposed to the fluctuating thermal state of the ambient environment. Effect of wind and relative humidity also play a role in affecting the temperature of the cask components. However, the most significant effects are the large thermal inertia of the system and the relatively low heat transfer coefficients that act to smooth out the daily temperature cycles. As a result, the amplitude of the cyclic stresses, to the extent that they are developed, remains orders of magnitude below the cask material's Endurance Limit.

The second causative factor, namely, pressure pulsation, is limited to the only pressure vessel in the system – the MPC. Pressure produces several types of stresses in the MPC (see Table 3.1.10), all of which are equally effective in causing fatigue expenditure in the metal. However, the amplitude of stress from the pressure cycling (due to the changes in the ambient conditions) is quite small and well below the endurance limit of the stainless steel material.

Therefore, failure from fatigue is not a credible concern for the HI-STORM FW system components.

### **3.1.2.6 Buckling**

Buckling is caused by a compressive stress acting on a slender section. In the HI-STORM FW system, the steel weldment in the overpack is not slender; its height-to-diameter ratio being less than 2. There is no source of compressive stress except from the self-weight of the shell and the overpack weight of the HI-TRAC VW in the stacked condition, which produces a modest state of compressive stress. The state of a small compressive stress combined with a low slenderness ratio makes the HI-STORM FW overpack safe from the buckling mode of failure. The same statement also applies to the HI-TRAC VW transfer cask, which is a radially buttressed triple shell (in comparison to the dual shell construction in HI-STORM FW) structure.

The MPC Enclosure Vessel is protected from buckling of by the permanent tensile stress in both hoop and longitudinal directions due to internal pressure.

Finally, the fuel basket, which is an egg-crate structure, as shown in Figures 1.1.6 and 1.1.7 (an intrinsically resistant structural form to buckling from axial compressive loads), is subject to minor compressive stresses from its own weight. The absence of buckling in the Metamic-HT fuel basket is based on the fact that there are no causative scenarios (normal or accident) that produce a significant in-plane compressive stress in the basket structure. A lower bound Euler Buckling strength for the Metamic-HT fuel basket can be obtained by assuming that the basket walls are fully continuous<sup>1</sup> over the entire height of the MPC fuel basket, neglecting the strengthening effect of the honeycomb completely, and treating the Metamic-HT basket wall as an end-loaded plate 199.5" high by 8.94" wide by 0.59" thick (corresponding to the maximum height MPC-37 fuel basket). The top and

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<sup>1</sup> In reality, the basket walls are not fully continuous in the vertical direction since the fuel basket is assembled by vertically stacking narrow width Metamic-HT panels in a honeycomb pattern (see drawing 6506 in Chapter 1 of HI-STORM FW SAR). For the above buckling strength evaluation, the assumption that the basket walls are continuous over the full height of the fuel basket is extremely conservative since the critical buckling load is inversely proportional to the square of the height.

bottom edges are assumed to be pinned and the lateral edges are assumed to be free to minimize the permissible buckling load (a particularly severe modeling artifice to minimize buckling strength). The Euler buckling load for this geometry is given by (see Timoshenko et al., “Theory of Elastic Stability”, 2<sup>nd</sup> Edition):

$$P_{cr} = \frac{\pi^2 EI}{h^2} = 125.2 lbf$$

where  $E$  = Young’s Modulus of Metamic-HT at 500°C = 3,300 ksi,  
 $I$  = moment of inertia of 8.94” wide by 0.59” thick plate = 0.153 in<sup>4</sup>,  
 $h$  = maximum height of fuel basket = 199.5”

The corresponding compressive axial stress is given by:

$$\sigma_{cr} = \frac{P_{cr}}{A} = \frac{125.2 lbf}{(8.94 in)(0.59 in)} = 23.7 psi$$

The factor of safety against buckling is given by (where  $\sigma_b$  is the compressive stress in the basket due to self weight):

$$SF = \frac{\sigma_{cr}}{\sigma_b} = \frac{23.7 psi}{19.5 psi} = 1.21$$

Thus, even with an exceedingly conservative model, the safety margin against buckling is more than 20%.

Therefore, buckling is ruled out as a credible failure mechanism in the HI-STORM FW system components. Nevertheless, a Design Basis Load consisting of external pressure is specified in Table 2.2.1 with the (evidently, non-mechanistic) conservative assumption that the internal pressure, which will counteract buckling behavior, is zero psig. (In reality, internal pressure cannot be zero because of the positive helium fill pressure established at the time of canister backfill.)

### 3.1.2.7 Consideration of Manufacturing and Material Deviations

Departure from the assumed values of material properties in the safety analyses clearly can have a significant effect on the computed margins. Likewise, the presence of deviations in manufacturing that inevitably occur in custom fabrication of capital equipment may detract from the safety factors reported in this chapter. In what follows, the method and measures adopted to insure that deviations in material properties or in the fabricated hardware will not undermine the structural safety conclusions are summarized.

That the yield and ultimate strengths of materials used in manufacturing the HI-STORM FW components will be greater than that assumed in the structural analyses is insured by the requirement

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in the ASME Code which mandates all Code materials to meet the minimum certified property values set down in the Code tables. Holtec International requires the material supplier to provide a Certified Mill Test Report in the format specified in the Code to insure compliance of all physical properties of the supplied material with the specified Code minimums. The same protocol to insure that the actual property values are above the minimum specified values is followed in the manufacture of Metamic-HT (Section 1.2.1.4.1 and Subsection 10.1.3). An additional margin in the actual physical properties vis-à-vis the Code values exists in the case of the MPC Confinement Boundary material by virtue of the Alloy X definition (Appendix 1.A): The physical properties of Alloy X at each temperature are set down at the lowest of that property value in the Code from a group of austenitic stainless steels.

The above measures make the probability of an actual material strength property to be falling below the assumed value in the structural analysis in this chapter to be non-credible. On the contrary, Holtec's manufacturing experience suggests that the actual properties are likely to be uniformly and substantially greater than the assumed values.

A similarly conservative approach is used to insure that the fabrication processes do not degrade the computed safety margins. Towards this end, the fabrication documents (drawings, travelers and shop procedures) implement a number of pro-active measures to prevent all known sources of development of a strength-adverse condition, such as:

- i. All welding procedures are qualified to yield better physical properties than the Code minimums. All essential variables that affect weld quality are tightly controlled.
- ii. Only those craftsmen who have passed the welding skill criteria implemented in the shop are permitted to weld.
- iii. A rigorous weld material quality overcheck program is employed to insure that every weld wire spool meets its respective Code specification.
- iv. All welds are specified as minimums: In practice, most exceed the specified minimums significantly. All primary structural welds are subject to Q.C. overcheck and sign-off.
- v. The Threaded Anchor Locations (TALs) are machined to a depth greater than the specified minimum. The stress analyses utilize the minimum thread depths/lengths per the licensing drawings.

In the event of a deviation that may depress the computed safety margin, a non-conformance report is prepared by the manufacturer and subject to a safety analysis by Holtec International's corporate engineering using the same methodology as that described in this FSAR. The item is accepted only if the safety evaluation musters part 72.48 acceptance criteria. A complete documentation of the life cycle of the NCR is archived in the Company's Permanent Filing System and shared with the designated system user.

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The above processes and measures have been in place at the Holtec Manufacturing Division to insure that an unacceptable reduction in the safety factors due to variation in material properties and manufacturing processes does not occur. The manufacturing experience over the past 20 years corroborates the effectiveness of the above measures.

### 3.1.3 Stress Analysis Models

To evaluate the effect of loads on the HI-STORM FW system components, finite element models for stress and deformation analysis are developed. The essential attributes of the finite element models for the HI-STORM overpack and the MPC are presented in this subsection. These models are used to perform the structural analysis of the system components under the loadings listed in Tables 2.2.6, 2.2.7 and 2.2.13, and summarized in Table 3.1.1 herein for handling, normal, off-normal, and accident conditions, respectively. The HI-TRAC VW transfer cask, on the other hand, is conservatively analyzed using strength of materials principles, as described in Subsection 3.1.3.3.

All finite element models are three-dimensional and are prepared to the level of discretization appropriate to the problem to be solved. The models are suitable for implementation in ANSYS and LS-DYNA general purpose codes, which are described in Subsection 3.6.2.

In the following, the finite element models of the HI-STORM overpack (body and lid) and the MPC (Confinement Boundary and the fuel baskets) are presented. Pursuant to ISG-21, the description of the computational model for each component addresses the following areas:

- Description of the model, its key attributes and its conservative aspects
- Types of finite elements used and the rationale for their selection
- Material properties and applicable temperature ranges
- Modeling simplifications and their underlying logic

In subsequent subsections, where the finite element models are deployed to analyze the different load cases, the presentation includes the consideration of:

- Geometric compliance of the simulation with the physics of the problem
- Boundary conditions
- Effect of tolerances on the results
- Convergence (numerical) of the solutions reported in this FSAR

The input files prepared to implement the finite element solutions as well as detailed results are archived in the Calculation Packages [3.4.11, 3.4.13] within the Company's Configuration Control

System. Essential portions of the results for each loading case necessary to draw safety conclusions are extracted from the Calculation Packages and reported in this FSAR. Specifically, the results summarized from the finite element solutions in this chapter are self-contained to enable an independent assessment of the system's safety. Input data is provided in tabular form as suggested in ISG-21. For consistency, the following units are employed to document input data throughout this chapter:

- Time: second
- Mass: pound
- Length: inch

### 3.1.3.1 HI-STORM FW Overpack

The physical geometry and materials of construction of the HI-STORM FW overpack are provided in Sections 1.1 and 1.2 and the drawings in Section 1.5. The finite element simulation of the overpack consists of two discrete models, one for the overpack body and the other for the top lid.

The models are initially developed using the finite element code ANSYS [3.4.1], and then, depending on the load case, numerical simulations are performed either in ANSYS or in LS-DYNA [3.1.8]. For example, the handling loads (Load Case 9) and the snow load (Load Case 10) are simulated in ANSYS, and the non-mechanistic tipover event (Load Case 4) is simulated in LS-DYNA. For the non-mechanistic tipover analysis, two distinct finite element models are created: one for the HI-STORM FW overpack carrying the maximum length MPC-37 and one for the HI-STORM FW overpack carrying the maximum length MPC-89 (Figures 3.4.10A and 3.4.10B).

The key attributes of the HI-STORM FW overpack models (implemented in ANSYS) are:

- i. The finite element discretization of the overpack is sufficiently detailed to accurately articulate the primary membrane and bending stresses as well as the secondary stresses at locations of gross structural discontinuity. The finite element layout of the HI-STORM FW overpack body and the top lid are pictorially illustrated in Figures 3.4.3 and 3.4.5, respectively. The overpack model consists of over 70,000 nodes and 50,000 elements, which exceed the number of nodes and elements in the HI-STORM 100 tipover model utilized in [3.1.4]. Table 3.1.11 summarizes the key input data that is used to create the finite element models of the HI-STORM FW overpack body and top lid.
- ii. The overpack baseplate, anchor blocks, and the lid studs are modeled with SOLID45 elements. The overpack inner and outer shells, bottom vent shells, and the lifting ribs are modeled with SHELL63 elements. A combination of SOLID45, SHELL63, and SOLSH190 elements is used to model the steel components in the HI-STORM FW lid. These element types are well suited for the overpack geometry and loading conditions, and they have been used successfully in previous cask licensing applications [3.1.10, 3.3.2].

- iii. All overpack steel members are represented by their linear elastic material properties (at 300°F) based on the data provided in Section 3.3. The concrete material in the overpack body is not explicitly modeled. Its mass, however, is accounted for by applying a uniformly distributed pressure on the baseplate annular area between the inner and outer shells (see Figure 3.4.26). The plain concrete in the HI-STORM FW lid is explicitly modeled in ANSYS using SOLID65 elements along with the input parameters listed in Table 3.1.12.
- iv. To implement the ANSYS finite element model in LS-DYNA, the SOLID45, SHELL63, and SOLSH190 elements are converted to solid, shell, and thick shell elements, respectively, in LS-DYNA. The SOLID65 elements used to model the plain concrete in the HI-STORM FW lid are replaced by MAT\_PSEUDO\_TENSOR (or MAT\_016) elements in LS-DYNA. The plain concrete in the overpack body is also modeled in LS-DYNA using MAT\_PSEUDO\_TENSOR elements.
- v. In LS-DYNA, all overpack steel members are represented by their applicable nonlinear elastic-plastic true stress-strain relationships. The methodology used for obtaining a true stress-strain curve from a set of engineering stress-strain data (e.g., strength properties from [3.3.1]) is provided in [3.1.9], which utilizes the following power law relation to represent the flow curve of metal in the plastic deformation region:

$$\sigma = K\varepsilon^n$$

where  $n$  is the strain-hardening exponent and  $K$  is the strength coefficient. Table 3.1.13 provides the values of  $K$  and  $n$  that are used to model the behavior of the overpack steel materials in LS-DYNA. Further details of the development of the true stress-strain relations for these materials are found in [3.4.11]. The concrete material is modeled in LS-DYNA using a non-linear material model (i.e., MAT\_PSEUDO\_TENSOR or MAT\_016) based on the properties listed in Section 3.3.

### 3.1.3.2 Multi-Purpose Canister (MPC)

The two constituent parts of the MPC, namely (i) the Enclosure Vessel and (ii) the Fuel Basket, are modeled separately. The model for the Enclosure Vessel is focused to quantify its stress and strain field under the various loading conditions. The model for the Fuel Basket is focused on characterizing its strain and displacement behavior during a non-mechanistic tipover event. For the non-mechanistic tipover analysis, two distinct finite element models are created: one for the maximum length MPC-37 and one for the maximum length MPC-89 (Figures 3.4.11 and 3.4.12).

The key attributes of the MPC finite element models (implemented in ANSYS) are:

- i. The finite element layout of the Enclosure Vessel is pictorially illustrated in Figure 3.4.1. The finite element discretization of the Enclosure Vessel is sufficiently detailed to accurately articulate the primary membrane and bending stresses as well as the secondary stresses at

locations of gross structural discontinuity, particularly at the MPC shell to baseplate juncture. This has been confirmed by comparing the ANSYS stress results with the analytical solution provided in [3.4.16] (specifically Cases 4a and 4b of Table 31) for the discontinuity stress at the junction between a cylindrical shell and a flat circular plate under internal pressure (100 psig). The two solutions agree within 3% indicating that the finite element mesh for the Enclosure Vessel is adequately sized. Table 3.1.14 summarizes the key input data that is used to create the finite element model of the Enclosure Vessel.

- ii. The Enclosure Vessel shell, baseplate, and upper and lower lids are meshed using SOLID185 elements. The MPC lid-to-shell weld and the reinforcing fillet weld at the shell-to-baseplate juncture are also explicitly modeled using SOLID185 elements (see Figure 3.4.1).
- iii. Consistent with the drawings in Section 1.5, the MPC lid is modeled as two separate plates, which are joined together along their perimeter edge. The upper lid is conservatively modeled as 4.5" thick, which is less than the minimum thickness specified on the licensing drawing (see Section 1.5). "Surface-to-surface" contact is defined over the interior interface between the two lid plates using CONTA173 and TARGE170 contact elements.
- iv. The materials used to represent the Enclosure Vessel are assumed to be isotropic and are assigned linear elastic material properties based on the Alloy X material data provided in Section 3.3. The Young's modulus value varies throughout the model based on the applied temperature distribution, which is shown in Figure 3.4.27 and conservatively bounds the normal operating temperature distribution for the maximum length MPC-37 as determined by the thermal analyses in Chapter 4.
- v. The fuel basket models (Figures 3.4.12A and 3.4.12B), which are implemented in LS-DYNA, are assembled from intersecting plates per the licensing drawings in Section 1.5, include all potential contacts and allow for relative rotations between intersecting plates. For conservatism, a bounding gap is assumed at contact interfaces between any two perpendicular basket plates to allow for impacts and, therefore, maximize the stress and deformation of the fuel basket plate. The fuel basket plates are modeled in LS-DYNA using thick shell elements, which behave like solid elements in contact, but can also accurately simulate the bending behavior of the fuel basket plates. To ensure numerical accuracy, full integration thick shell elements with 10 through-thickness integration points are used. This modeling approach is consistent with the approach taken in [3.1.10] to qualify the F-32 and F-37 fuel baskets.
- vi. In LS-DYNA, the fuel basket plates are represented by their applicable nonlinear elastic-plastic true stress-strain relationships in the same manner as the steel members of the HI-STORM FW overpack (see Subsection 3.1.3.1). Table 3.1.13 provides the values of K and n that are used to model the behavior of the fuel basket plates in LS-DYNA. Details of the development of the true stress-strain relations are found in [3.4.11].

### 3.1.3.3 HI-TRAC VW Transfer Cask

The stress analysis of the transfer cask addresses three performance features that are of safety consequence. They are:

- i. Performance of the water jacket as a pressure retaining enclosure under an accident condition leading to overheating of water.
- ii. Performance of the threaded anchor locations in the HI-TRAC VW top flange under the maximum lifted load.
- iii. Performance of the HI-TRAC VW bottom lid under its own self weight plus the weight of the heaviest MPC.

The above HI-TRAC VW components are analyzed separately using strength of materials formula, the details of which are provided in Subsections 3.4.3 and 3.4.4.



Table 3.1.1

## GOVERNING CASES AND AFFECTED COMPONENTS

Case	Loading Case I.D. from Tables 2.2.6, 2.2.7 and 2.2.13	Loading Event	Affected Components			Objective of the Analysis	For additional discussion, refer to Subsection
			HI-STORM	MPC	HI-TRAC		
1	AD	<u>Moving Flood</u> Moving Floodwater with loaded HI-STORM on the pad.	X	—	—	Determine the flood velocity that will not overturn the overpack.	2.2.3
2.	AE	<u>Design Basis Earthquake (DBE)</u> Loaded HI-STORMs arrayed on the ISFSI pad subject to ISFSI's DBE	X	X	—	Determine the maximum magnitude of the earthquake that meets the acceptance criteria of 2.2.3(g).	2.2.3
3	AC	<u>Tornado Missile</u> A large, medium or small tornado missile strikes a loaded HI-STORM on the ISFSI pad or HI-TRAC.	X	X	X	Demonstrate that the acceptance criteria of 2.2.3(e) will be met.	2.2.3
4	AA	<u>Non-Mechanistic Tip-Over</u> A loaded HI-STORM is assumed to tip over and strike the pad.	X	X	—	Satisfy the acceptance criteria of 2.2.3(b).	2.2.3
5	NB	<u>Design Internal Pressure</u> MPC under the normal condition Design Internal Pressure	—	X	—	Demonstrate that the MPC meets "NB" stress intensity limits.	2.2.1
6	NB	<u>Maximum Internal Pressure Under the Accident Condition</u> MPC under the accident condition internal pressure (from Table 2.2.1)	—	X	—	Demonstrate that the Level D stress intensity limits are met.	2.2.1

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Table 3.1.1 (continued)

## GOVERNING CASES AND AFFECTED COMPONENTS

Case	Loading Case I.D. from Tables 2.2.6, 2.2.7 and 2.2.13	Loading Event	Affected Components			Objective of the Analysis	For additional discussion, refer to Subsection
7	AH	<u>Design External Pressure</u> MPC under the accident condition external pressure (from Table 2.2.1)	—	X	—	The Enclosure Vessel must not buckle.	2.2.3
8	AJ	<u>HI-TRAC Non-Mechanistic Heat-Up</u> Postulate the water jacket's internal pressure reaches the Design Pressure (defined in Table 2.2.1)	—	—	X	Demonstrate that the stresses in the water jacket meet the ASME Code Section III Subsection Class 3 limits for the Design Condition.	2.2.1
9.	HA, HB, and HC	<u>Handling of Components</u>	X	X	X	Demonstrate that the tapped anchor locations (TALs) meet the Regulatory Guide 3.61 and NUREG-0612 stress limits (as applicable).	2.2.1
10.	NA	<u>Snow Load</u>	X	—	—	Demonstrate that the top lid's steel structure meets "NF" stress limit for normal condition.	2.2.1
11.	NA	<u>MPC Reflood Event</u>	—	X	—	Demonstrate that there is no breach of the fuel rod cladding.	12.3.1

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Table 3.1.2

## DESIGN AND LEVEL A: STRESS

Reference Code: ASME NF  
 Material: SA36  
 Service Conditions: Design and Normal  
 Item: Stress

Temp. (Deg. F)	Classification and Value (ksi)		
	S	Membrane Stress	Membrane plus Bending Stress
-20 to 650	16.6	16.6	24.9
700	15.6	15.6	23.4

## 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 3.1.3

## LEVEL B: STRESS

Reference Code: ASME NF  
 Material: SA36  
 Service Conditions: Off-Normal  
 Item: Stress

Temp. (Deg. F)	Classification and Value (ksi)	
	Membrane Stress	Membrane plus Bending Stress
-20 to 650	22.1	33.1
700	20.7	31.1

Notes:

1. Limits on values are presented in Table 2.2.12 with allowables from Table 3.1.2.

Table 3.1.4

## DESIGN AND LEVEL A SERVICE CONDITIONS: ALLOWABLE STRESS

Code: ASME NF  
 Material: SA516 (SA515) Grade 70, SA350-LF3 (SA350-LF2)  
 Service Conditions: Design and Normal  
 Item: Allowable Stress

Temp. (Deg. F)	Classification and Value (ksi)		
	S	Membrane Stress	Membrane plus Bending Stress
-20 to 400	20.0	20.0	30.0
500	19.6	19.6	29.4
600	18.4	18.4	27.6
650	17.8	17.8	26.7
700	17.2	17.2	25.8

## 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.
4. Maximum allowable stress values are the lowest of all values for the candidate materials (SA516 (SA515) Grade 70, SA350-LF3 (SA350-LF2)) at corresponding temperature.

Table 3.1.5

## LEVEL B: ALLOWABLE STRESS

Code: ASME NF  
 Material: SA516 (SA515) Grade 70, SA350-LF3 (SA350-LF2)  
 Service Conditions: Off-Normal  
 Item: Allowable Stress

Temp. (Deg. F)	Classification and Value (ksi)	
	Membrane Stress	Membrane plus Bending Stress
-20 to 400	26.6	39.9
500	26.1	39.1
600	24.5	36.7
650	23.7	35.5
700	22.9	34.3

## Notes:

1. Limits on values are presented in Table 2.2.12 with allowables from Table 3.1.4.
2. Maximum allowable stress values are the lowest of all values for the candidate materials (SA516 (SA515) Grade 70, SA350-LF3 (SA350-LF2)) at corresponding temperature.

Table 3.1.6

## LEVEL D: STRESS INTENSITY

Code: ASME NF  
 Material: SA516 (SA515) Grade 70  
 Service Conditions: Accident  
 Item: Stress Intensity

Temp. (Deg. F)	Classification and Value (ksi)		
	$S_m$	$P_m$ AMAX ( $1.2S_y$ , $1.5S_m$ ), but $< 0.7 S_u$	$P_m + P_b$ 150% of $P_m$
-20 to 100	23.3	45.6	68.4
200	23.2	41.8	62.7
300	22.4	40.3	60.4
400	21.6	39.0	58.5
500	20.6	37.2	55.8
600	19.4	34.9	52.4
650	18.8	33.8	50.7
700	18.1	32.9	49.4

## 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.  $P_m$  and  $P_b$  are defined in Table 3.1.10.

Table 3.1.7

## DESIGN, LEVELS A AND B: STRESS INTENSITY

Code: ASME NB  
 Material: Alloy X  
 Service Conditions: Design, Levels A and B (Normal and Off-Normal)  
 Item: Stress Intensity

Temp. (Deg. F)	Classification and Numerical Value					
	$S_m$	$P_m^\dagger$	$P_L^\dagger$	$P_L + P_b^\dagger$	$P_L + P_b + Q^{\dagger\dagger}$	$P_e^{\dagger\dagger}$
-20 to 100	20.0	20.0	30.0	30.0	60.0	60.0
200	20.0	20.0	30.0	30.0	60.0	60.0
300	20.0	20.0	30.0	30.0	60.0	60.0
400	18.6	18.6	27.9	27.9	55.8	55.8
500	17.5	17.5	26.3	26.3	52.5	52.5
600	16.5	16.5	24.75	24.75	49.5	49.5
650	16.0	16.0	24.0	24.0	48.0	48.0
700	15.6	15.6	23.4	23.4	46.8	46.8
750	15.2	15.2	22.8	22.8	45.6	45.6
800	14.8	14.8	22.2	22.2	44.4	44.4

## 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 corresponding 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 3.1.10.

$^\dagger$  Evaluation required for Design condition only.

$^{\dagger\dagger}$  Evaluation required for Levels A, B conditions only.  $P_e$  not applicable to vessels.

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Table 3.1.8

## LEVEL D: STRESS INTENSITY

Code: ASME NB  
 Material: Alloy X  
 Service Conditions: Level D (Accident)  
 Item: Stress Intensity

Temp. (Deg. F)	Classification and Value (ksi)		
	P <sub>m</sub>	P <sub>L</sub>	P <sub>L</sub> + P <sub>b</sub>
-20 to 100	48.0	72.0	72.0
200	48.0	72.0	72.0
300	46.3	69.45	69.45
400	44.6	66.9	66.9
500	42.0	63.0	63.0
600	39.6	59.4	59.4
650	38.4	57.6	57.6
700	37.4	56.1	56.1
750	36.5	54.8	54.8
800	35.5	53.25	53.25

## 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<sub>u</sub>.
3. Limits on values are presented in Table 2.2.10.
4. P<sub>m</sub>, P<sub>L</sub>, and P<sub>b</sub> are defined in Table 3.1.10.

Table 3.1.9

## FRACTURE TOUGHNESS TEST REQUIREMENTS FOR HI-STORM FW OVERPACK

Material	Test Requirement	Test Temperature	Acceptance Criterion
Bolting (SA193 B7)	Not required per NF-2311(b)(13) and Note (e) to Figure NF-2311(b)-1	-	-
Material with a nominal section thickness of 5/8" and less	Not required per NF-2311(b)(1)	-	-
Normalized SA516 Gr. 70 (thicknesses 2-1/2" and less)	Not required per NF-2311(b)(10) for service temperatures greater than or equal to 0°F (i.e., handling operations), and per NF-2311(b)(7) for service temperatures less than 0°F and greater than or equal to -40°F (i.e., non-handling operations)	-	-
Normalized SA516 Gr. 70 used for HI-STORM FW base plate (thickness greater than 2-1/2")	Not required per NF-2311(b)(7)	-	-
As rolled SA516 Gr. 70 used for HI-STORM FW inner and outer shells, base plate, top plate, inlet shell plate, inlet vent top plate, gamma shield plate, lid lower shim plate and lid gusset	Not required per NF-2311(b)(7)	-	-
SA36 (thickness greater than 5/8")	Not required per NF-2311(b)(7)	-	-
SA350-LF2 (thickness greater than 5/8") and as rolled SA516 Gr. 70 used for HI-STORM FW lifting rib	Per NF-2331	-40°F (Also must meet ASME Section IIA requirements)	Table NF-2331(a)-3 or Figure NF-2331(a)-2 (Also must meet ASME Section IIA requirements)
Weld material	Test per NF-2430 if: (1) either of the base materials of the production weld requires impact testing, or; (2) either of the base materials is SA516 Gr. 70 with nominal section thickness greater than 5/8".	-40°F	Per NF-2331

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Table 3.1.9 (continued)

## FRACTURE TOUGHNESS TEST REQUIREMENTS FOR HI-TRAC VW TRANSFER CASK

Material	Test Requirement	Test Temperature	Acceptance Criterion
Bolting (SA193 B7)	Not required per NF-2311(b)(13) and Note (e) to Figure NF-2311(b)-1	-	-
Bolting (SA564 630 H1150)	Per NF-2333	0°F (Also must meet ASME Section II.A requirements)	Table NF-2333-1 (Also must meet ASME Section II.A requirements)
Material with a nominal section thickness of 5/8" and less	Not required per NF-2311(b)(1)	-	-
Normalized SA516 Gr. 70 (thicknesses 2-1/2" and less)	Not required per NF-2311(b)(10)	-	-
Normalized SA516 Gr. 70 used for HI-TRAC VW bottom lid (thickness greater than 2-1/2")	Not required per NF-2311(b)(7)	-	-
As rolled SA516 Gr. 70 used for HI-TRAC VW inner and outer shells, bottom flange, extended rib, short rib, bolt recess cap, and bottom lid	Not required per NF-2311(b)(7)	-	-
SA36 (thickness greater than 5/8")	Not required per NF-2311(b)(7)	-	-
SA515 Gr. 70, SA106 Gr. C, and SA350-LF3 (thickness greater than 5/8")	Per NF-2331	0°F (Also must meet ASME Section IIA requirements)	Table NF-2331(a)-3 or Figure NF-2331(a)-2 (Also must meet ASME Section IIA requirements)
Weld material	Test per NF-2430 if: (1) either of the base materials of the production weld requires impact testing, or; (2) either of the base materials is SA516 Gr. 70 with nominal section thickness greater than 5/8".	0°F	Per NF-2331

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Table 3.1.10

## ORIGIN, TYPE AND SIGNIFICANCE OF STRESSES IN THE HI-STORM FW SYSEM

Symbol	Description	Notes
$P_m$	Primary membrane stress	Excludes effects of discontinuities and concentrations. Produced by pressure and mechanical loads. Primary membrane stress develops in the MPC Enclosure Vessel shell. Limits on $P_m$ exist for normal (Level A), off-normal (Level B), and accident (Level D) service conditions.
$P_L$	Local membrane stress	Considers effects of discontinuities but not concentrations. Produced by pressure and mechanical loads, including earthquake inertial effects. $P_L$ develops in the MPC Enclosure Vessel wall due to impact between the overpack guide tubes and the MPC (near the top of the MPC) under an earthquake (Level D condition) or non-mechanistic tip-over event. However, because there is no Code limit on $P_L$ under Level D event, a limit on the local strain consistent with the approach in the HI-STORM 100 docket is used (see Subsection 3.4.4.1.4).
$P_b$	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 earthquake inertial effects. Primary bending stress develops in the top lid and baseplate of the MPC, which is a pressurized vessel. Lifting of the loaded MPC using the so-called "lift cleats" also produces primary bending stress in the MPC lid. Similarly, the top lid of the HI-STORM FW module, a plate-type structure, withstands the snow load (Table 2.2.8) by developing primary bending stress.
$P_e$	Secondary expansion stress	Stresses that result from the constraint of free-end displacement. Considers effects of discontinuities but not local stress concentration (not applicable to vessels). 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-STORM FW component.
$Q$	Secondary membrane plus bending stress	Self-equilibrating stress necessary to satisfy continuity of structure. Occurs at gross structural discontinuities. Can be caused by pressure, mechanical loads, or differential thermal expansion. The junction of MPC shell with the baseplate and top lid locations of gross structural discontinuity, where secondary stresses develop as a result of internal pressure. Secondary stresses would also develop at the two extremities of the MPC shell if a thermal gradient were to exist. However, because the top and bottom regions of the MPC cavity also serve as the top and bottom plenums, respectively, for the recirculating helium, the temperature field in the regions of gross discontinuity is essentially uniform, and as a result, the thermal stress adder is insignificant and neglected (see Paragraph 3.1.2.5).
$F$	Peak stress	Increment added to primary or secondary stress by a concentration (notch), or, certain thermal stresses that may cause fatigue but not distortion. Because fatigue is not a credible source of failure in a passive system with gradual temperature changes, fatigue damage is not computed for HI-STORM FW components.

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Table 3.1.11

KEY INPUT DATA FOR FINITE ELEMENT MODEL OF HI-STORM FW OVERPACK	
Item	Value
Overall height of HI-STORM FW (including top lid)	221.5 in (for maximum length BWR fuel) 239.5 in (for maximum length PWR fuel)
Height of overpack body	199.25 (for maximum length BWR fuel) 217.25 in (for maximum length PWR fuel)
Height of top lid above top of overpack body	22.25 in
Top lid diameter	103 in
Inside diameter of HI-STORM FW storage cavity	81 in
Outside diameter of HI-STORM FW overpack	139 in
Inner shell thickness	0.75 in
Outer shell thickness	0.75 in
Lifting rib thickness	1 in
Baseplate thickness	3 in
Material	Various (see licensing drawings in Section 1.5)
Ref. temperature for material properties	300°F (implemented in ANSYS) Table 3.1.13 (implemented in LS-DYNA)
Concrete density	200 lbf/ft <sup>3</sup>

Table 3.1.12

INPUT PARAMETERS FOR SOLID65 CONCRETE ELEMENTS  
USED IN HI-STORM FW LID MODEL

Input Parameter	Value
Density	200 lbf/ft <sup>3</sup>
Poisson's ratio	0.17
Compressive strength	3,000 psi
Young's modulus	$3.122 \times 10^6$ psi
Shear transfer coefficient for open cracks	0.1
Shear transfer coefficient for closed cracks	0.3

Table 3.1.13

VALUES OF “K” AND “n” USED TO MODEL ELASTIC-PLASTIC BEHAVIOR  
OF HI-STORM SYSTEM COMPONENTS IN LS-DYNA

Component	Material	Ref. Temperature	$K^{\dagger}$ (psi)	$n^{\dagger}$
Fuel Basket	Metamic-HT	365°C	$1.421 \times 10^4$	0.059
		350°C	$1.506 \times 10^4$	0.062
		325°C	$1.705 \times 10^4$	0.055
		300°C	$1.901 \times 10^4$	0.049
		250°C	$2.184 \times 10^4$	0.064
		200°C	$2.461 \times 10^4$	0.075
MPC Lid	Alloy X	500°F	$1.055 \times 10^5$	0.235
MPC Shell	Alloy X	450°F	$1.152 \times 10^5$	0.244
MPC Baseplate	Alloy X	350°F	$1.161 \times 10^5$	0.236
HI-STORM Anchor Block	SA-350 LF2	250°F	$1.160 \times 10^5$	0.189
HI-STORM Lid Stud	SA-193 B7	250°F	$1.399 \times 10^5$	0.082
HI-STORM Inlet Shield Pipe	SA-53	250°F	$9.464 \times 10^4$	0.161
HI-STORM Body <sup>††</sup>	SA-516 Gr. 70	300°F	$1.144 \times 10^5$	0.181
HI-STORM Lid	SA-516 Gr. 70	250°F	$1.139 \times 10^5$	0.179
HI-STORM Inlet Shell Plate, Inlet Vent Top Plate, & Lid	SA-36	250°F	$8.952 \times 10^4$	0.150

<sup>†</sup> K and n are defined in Subsection 3.1.3.1.

<sup>††</sup> Includes all components in HI-STORM overpack body made from SA-516 Gr. 70 material (e.g., baseplate, inner and outer shells, lifting ribs, etc.).

Table 3.1.14

KEY INPUT DATA FOR ANSYS MODEL OF MPC ENCLOSURE VESSEL	
Item	Value
Overall Height of MPC	195 in (for maximum length BWR fuel) 213 in (for maximum length PWR fuel)
Outside diameter of MPC	75.5 in
MPC upper lid thickness	4.5 in
MPC lower lid thickness	4.5 in
MPC shell thickness	0.5 in
MPC baseplate thickness	3.0 in
Material	Alloy X
Ref. temperature for material properties	Figure 3.4.27 (implemented in ANSYS) Table 3.1.13 (implemented in LS-DYNA)



## 3.2 WEIGHTS AND CENTERS OF GRAVITY

As stated in Chapter 1, while the diameters of the MPC, HI-STORM FW, and HI-TRAC VW are fixed, their height is dependent on the length of the fuel assembly. The MPC cavity height (which determines the external height of the MPC) is set equal to the nominal fuel length (along with control components, if any) plus  $\Delta$ , where  $\Delta$  is between 1.5" (minimum), 2.0" (maximum),  $\Delta$  is increased above 1.5" so that the MPC cavity height is a full inch or half-inch number. Thus, for the PWR reference fuel (Table 1.0.4), whose length including control components is 167.2" (Table 2.1.1),  $\Delta = 1.8$ " so that the MPC cavity height,  $c$ , becomes 169".  $\Delta$  is provided to account for irradiation and thermal growth of the fuel in the reactor. Table 3.2.1 provides the height of the internal cavities and bottom-to-top external dimension of all system components. Table 3.2.2 provides the parameters that affect the weight of cask components and their range of values assumed in this FSAR.

The cavity heights of the HI-STORM FW overpack and the HI-TRAC VW transfer cask are set greater than the MPC height by fixed amounts to account for differential thermal expansion and manufacturing tolerances. Table 3.2.1 provides the height data on HI-STORM FW, HI-TRAC VW, and the MPC as the adder to the MPC cavity length.

Table 3.2.5 provides the reference weight of the HI-STORM FW overpack for storing MPC-37 and MPC-89 containing reference PWR and BWR fuel, respectively. The weight of the HI-STORM FW overpack body is provided for two discrete concrete densities and for two discrete heights for PWR and BWR fuel. The weight at any other density and any other height can be obtained by linear interpolation. Similarly the weight of the HI-STORM FW lid is provided for two discrete values of concrete density. The weight corresponding to any other density can be computed by linear interpolation.

As discussed in Section 1.2, the weight of the HI-TRAC VW transfer cask is maximized for a particular site to take full advantage of the plant's crane capacity within the architectural limitations of the Fuel Building. Accordingly, the thickness of the lead shield and outer diameter of the water jacket can be increased to maximize shielding. The weight of the empty HI-TRAC VW cask in Table 3.2.4 is provided for three lengths corresponding to PWR fuel. Using the data for three lengths, the transfer cask's weight corresponding to any other length can be obtained by linear interpolation (or extrapolation). For MPC-89, the weight data is provided for the minimum and reference fuel lengths, as well as the reference fuel assembly with a DFC and therefore likewise the transfer cask's weight corresponding to any other length can be obtained by linear interpolation (or extrapolation).

The approximate change in the empty weight of HI-TRAC VW (in kilo pounds) of a certain height,  $h$  (inch), by virtue of changing the thickness of the lead by an amount,  $\delta$  (inch), is given by the formula:

$$\Delta W_{lead} = 0.1128(h - 13.5) \delta$$

The approximate change in the empty weight of HI-TRAC VW (in kilo pounds) of a certain length,  $h$  (inch), by virtue of changing the thickness of the water layer by  $\delta$  (inch) is given by:

$$\Delta W_{water} = 0.01077 (h - 13.5) \delta$$

The above formulas serve as a reasonable approximation for the weight change whether the thickness of lead (or water) is being increased or decreased.

The weights of the loaded MPCs containing “reference SNF” with and without water are provided in Table 3.2.3. All weights in the aforementioned tables are nominal values computed using the SOLIDWORKS™ computer code or using standard material density and geometric shapes for the respective subcomponents of the equipment.

Table 3.2.5 provides the loaded weight of the HI-STORM FW system on the ISFSI pad for two different concrete densities for both PWR and BWR reference fuel. Table 3.2.6 contains the weight data on loaded HI-TRAC VW under the various handling scenarios expected during loading.

The maximum and minimum locations of the centers of gravity (CGs) are presented (in dimensionless form) in Table 3.2.7. The radial eccentricity,  $\phi$ , of a cask system is defined as:

$$\phi = \frac{\Delta_r}{D} \times 100 \quad (\phi \text{ is dimensionless})$$

where  $\Delta_r$  is the radial offset distance between the CG of the cask system and the geometric centerline of the cask, and  $D$  is the outside diameter of the cask. In other words, the value of  $\phi$  defines a circle around the axis of symmetry of the cask within which the CG lies (see Figure 3.2.1). All centers of gravity are located close to the geometric centerline of the cylindrical cask since the non-axisymmetric effects of the cask system and its contents are very small. The vertical eccentricity,  $\Psi$ , of a cask system is defined similarly as:

$$\Psi = \frac{\Delta_v}{H} \times 100 \quad (\Psi \text{ is dimensionless})$$

Where  $\Delta_v$  is the vertical offset distance between the CG of the cask system and the geometric center of the cask (i.e., cask mid-height), and  $H$  is the overall height of the cask. A positive value of  $\Psi$  indicates that the CG is located above the cask mid-height, and a negative value indicates that the CG is located below the cask mid-height. Figure 3.2.2 illustrates how  $\Psi$  is defined.

The values of  $\phi$  and  $\Psi$  given in Table 3.2.7 are bounding values, which take into consideration material and fabrication tolerances. The tabulated values of  $\phi$  and  $\Psi$  can be converted into dimensionless form using the equations above. For example, from Table 3.2.7 the empty HI-STORM FW with lid installed has maximum eccentricities of  $\phi = 2.0$  and  $\Psi = \pm 3.0$ . Therefore, the maximum radial and vertical offset distances are ( $D=140''$ ,  $H=207.75''$  for PWR reference fuel):

$$\Delta_r = \frac{\phi D}{100} = \frac{(2.0)(140in)}{100} = 2.8in$$

$$\Delta_v = \frac{\Psi H}{100} = \frac{(\pm 3.0)(207.75in)}{100} = \pm 6.23in \text{ (CG height relative to H/2)}$$

The C.G. information provided above shall be used in designing the lifting and handling ancillary for the HI-STORM FW cask components. In addition, the maximum CG height per Table 3.2.7 shall be used for the stability analysis of the HI-STORM FW under DBE conditions. Using the weight data in the previously mentioned tables, Table 3.2.8 has been constructed to provide the bounding weights for structural analyses so that every load case is analyzed using the most conservative data (to *minimize the computed safety margins*). The weight data in Table 3.2.8 is used in all structural analyses in this chapter.

Table 3.2.1 OPTIMIZED MPC, HI-TRAC, AND HI-STORM HEIGHT DATA FOR A SPECIFIC UNIRRADIATED FUEL LENGTH, $\ell^{\dagger}$	
MPC Cavity Height, c	$\ell + \Delta^{\ddagger}$
MPC Height (including top lid, excluding closure ring), h	c + 12"
HI-TRAC VW Cavity Height	h + 1"
HI-TRAC VW Total Height	h + 6.5"
HI-STORM FW Cavity Height	h + 3.5"
HI-STORM FW Body Height (height from the bottom of the HI-STORM FW to the top surface of the shear ring at the top of the HI-STORM FW body)	h + 4.5"
HI-STORM FW Height (loaded over the pad)	h + 27"

<sup>†</sup> Fuel Length,  $\ell$ , shall be based on the fuel assembly length with or without a damaged fuel container (DFC). Users planning to store fuel in DFCs shall adjust the length  $\ell$  to include the additional height of the DFC. The maximum additional height for the DFC shall be 5". Note that users who plan to store any fuel in a DFC will need to utilize a system designed for the additional length and will need to use fuel shims (if required) to reduce the gap between the fuel without a DFC and the enclosure cavity to approximately 1.5-2.5 inches.

<sup>‡</sup>  $\Delta$  shall be selected as  $1.5'' < \Delta < 2.0''$  so that c is an integral multiple of 1/2 inch (add 1.5" to the fuel length and round up to the nearest 1/2" or full inch).

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Table 3.2.2			
LIMITING PARAMETERS			
	Item	PWR	BWR
1.	Minimum fuel assembly length, inch	150	171
2.	Maximum fuel assembly length, inch	199.2	181.5 <sup>3</sup>
3.	Nominal thickness of the lead cylinder in the lowest weight HI-TRAC VW, inch	2.75	2.50
4.	Maximum nominal thickness of the lead cylinder, inch	4.25	4.25
5.	Nominal (radial) thickness of the water in the external jacket, inch	4.75	4.75

<sup>3</sup> Maximum fuel assembly length for the BWR fuel assembly refers to the maximum fuel assembly length plus an additional 5" to account for a Damage Fuel Container (DFC).

Table 3.2.3						
MPC WEIGHT DATA (COMPUTED NOMINAL VALUES)						
Item	BWR Fuel Based on length below			PWR Fuel Based on length below		
	Reference	Shortest from Table 3.2.2	Longest from Table 3.2.2	Reference	Shortest from Table 3.2.2	Longest from Table 3.2.2
Enclosure Vessel	27,500	27,100	27,800	28,600	25,600	31,100
Fuel Basket	8,600	8,300	8,800	7,900	7,000	9,400
Water in the MPC @ SG = 1 (See Note 1)	16,700	16,200	18,900	15,400	14,000	18,700
Water mass displaced by a closed MPC Enclosure Vessel (SG = 1)	30,800	29,900	31,600	29,300	26,600	34,500

SG = Specific Gravity

Note 1: Water weight in the MPC assumes that water volume displaced by the fuel is equal to the fuel weight divided by an average fuel assembly density of 0.396 lb/in<sup>3</sup>. The fuel weights used for calculating the fuel volumes for Reference/Shortest/Longest PWR and BWR fuel assemblies are 1750/1450/2050 and 750/700/850 pounds respectively.

Table 3.2.4						
HI-TRAC VW WEIGHT DATA (COMPUTED NOMINAL VALUES)						
Item	BWR Fuel Based on length below			PWR Fuel Based on length below		
	Reference	Shortest from Table 3.2.2	Longest from Table 3.2.2	Reference	Shortest from Table 3.2.2	Longest from Table 3.2.2
HI-TRAC VW Body (no Bottom Lid, water jacket empty)	84,000	81,700	86,200	85,200	78,000	99,600
HI-TRAC VW Bottom Lid	11,300	11,300	11,300	11,300	11,300	11,300
MPC with Basket	36,100	35,400	36,600	36,500	32,600	40,500
Fuel Weight (assume 50% with control components or channels, as applicable)	66,800 (750 lb per assembly average)	64,600 (725 lb per assembly average)	71,200 (800 lb per assembly average)	62,000 (1,675 lb per assembly average)	53,700 (1,450 lb per assembly average)	69,400 (1,875 lb per assembly average)
Water in the Annulus	600	600	600	600	600	700
Water in the Water Jacket	8,800	8,500	9,000	8,400	7,600	9,900
Displaced Water Mass by the Cask in the Pool (Excludes MPC)	18,900	18,400	19,400	18,600	17,500	21,600

Table 3.2.5

## ON-ISFSI WEIGHT OF LOADED HI-STORM FW

Scenario		Weight of Cask Body (kilo-pounds)	Weight of HI-STORM FW Lid (kilo-pounds)
Fuel Type	HI-STORM FW Concrete Density (lb/cubic feet)		
Ref. PWR	150	198.0	20.1
Ref. PWR	200	246.2	23.3
Maximum length – PWR	150	229.0	20.1
Maximum length – PWR	200	286.1	23.3
Ref. BWR	150	206.7	20.1
Ref. BWR	200	257.4	23.3
Maximum length – BWR	150	211.6	20.1
Maximum length – BWR	200	263.7	23.3

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Table 3.2.6				
HI-TRAC VW OPERATING WEIGHT DATA FOR REFERENCE FUEL				
Scenario			HI-TRAC VW <sup>4</sup> Weight in Kilo-Pounds	
Water in the MPC	Water in the Water Jacket	Cask in (pool) Water/Air	Ref. PWR Fuel	Ref. BWR Fuel
Yes	Yes	Water	167.7	173.3
Yes	Yes	Air	215.5	222.9
Yes	No	Water	159.4	164.6
No	No	Water	143.7	147.9
No	Yes	Air	199.9	206.2
No	No	Air	191.5	197.5

Weights above include the weight of the fuel assembly alone and do not include any additional weight for non-fuel hardware or damaged fuel containers.

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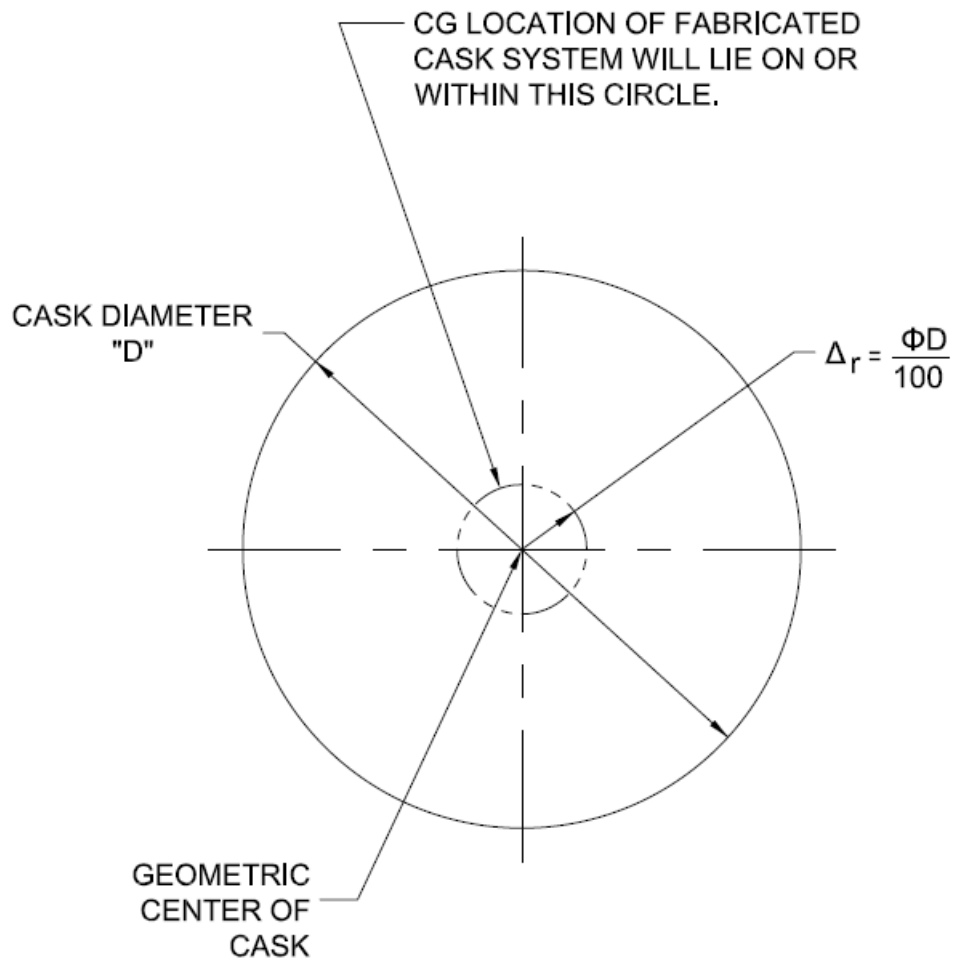
<sup>4</sup> Add 4,000 lbs for the weight of the lift yoke.

Table 3.2.7

LOCATION OF C.G. WITH RESPECT TO THE CENTERPOINT ON THE EQUIPMENT'S GEOMETRIC CENTERLINE			
	Item	Radial eccentricity (dimensionless) <sup>5</sup> , $\phi$	Vertical eccentricity (dimensionless), Above (+)* or Below (-), $\psi$
1.	Empty HI-STORM FW with lid installed	2.0	$\pm 3.0$
2.	Empty HI-STORM FW without top lid	2.0	$\pm 3.0$
3.	HI-STORM FW with fully loaded stored MPC without top lid	2.0	$\pm 2.0$
4.	HI-STORM FW with lid and a fully loaded MPC	2.0	$\pm 3.0$
5.	HI-TRAC VW with Bottom lid and loaded MPC	2.0	$\pm 2.0$
6.	Empty HI-TRAC VW without bottom lid	2.0	$\pm 2.0$

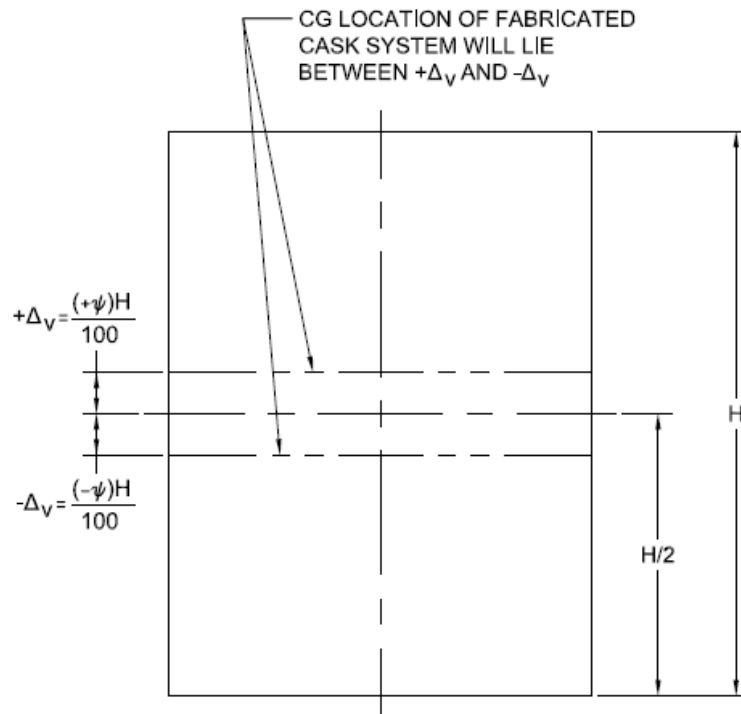
<sup>5</sup>  $\phi$  and  $\Psi$  are dimension values as explained in Section 3.2.

Table 3.2.8			
BOUNDING WEIGHTS FOR STRUCTURAL ANALYSES (Height from Tables 3.2.1 and 3.2.2)			
	Case	Purpose	Assumed Weight (Kilo-pounds)
1.	Loaded HI-STORM FW on the pad containing maximum length/weight fuel and 200 lb/cubic feet concrete – maximum possible weight scenario	Sizing and analysis of lifting and handling locations and cask stability analysis under overturning loads such as flood and earthquake	425.7
2.	Loaded HI-STORM FW on the pad with 150 lb concrete, shortest length MPC	Stability analysis under missile strike	285.7
3.	Loaded HI-TRAC VW with maximum length fuel and maximum lead and water shielding	Analysis for NUREG-0612 compliance of lifting and handling locations (TALs)	270.0
4.	Loaded HI-TRAC VW with shortest length MPC and minimum lead and water shielding	Stability analysis under missile strike	183.5
5.	Loaded MPC containing maximum length/weight fuel – maximum possible weight scenario	Analysis for NUREG-0612 compliance of lifting and handling locations (TALs)	116.4



Top View of Cask

Figure 3.2.1: Radial Eccentricity of Cask Center of Gravity



Elevation View of Cask

Figure 3.2.2: Vertical Eccentricity of Cask Center of Gravity

### 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 which envelopes the maximum and minimum temperatures under all service conditions applicable to the HI-STORM FW system components.

The materials selected for use in the MPC, HI-STORM FW overpack, and HI-TRAC VW transfer cask are presented on the drawings in Section 1.5. In this chapter, the materials are divided into two categories, structural and nonstructural. Structural materials are materials that act as load bearing members and are, therefore, significant in the stress evaluations. Materials that do not support mechanical loads are considered nonstructural. For example, the HI-TRAC VW inner shell is a structural material, while the lead between the inner and outer shell is a nonstructural material. For nonstructural materials, the principal property that is used in the structural analysis is weight density. In local deformation analysis, however, such as the study of penetration from a tornado-borne missile, the properties of lead in HI-TRAC VW and plain concrete in HI-STORM FW are included.

#### 3.3.1 Structural Materials

##### a. Alloy X

A hypothetical material termed Alloy X is defined for the MPC pressure retaining boundary. The material properties of Alloy X are the least favorable values from the set of candidate 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 will be of equal or greater strength than the hypothetical material used in the analysis.

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 in all structural analyses. As is shown in Chapter 4, the maximum metal temperature for Alloy X used at or within the Confinement Boundary remains below 1000°F under all service modes. As shown in ASME Code Case N-47-33 (Class 1 Components in Elevated Temperature Service, 2007 Code Cases, Nuclear Components), the strength properties of austenitic stainless steels do not change due to exposure to 1000°F temperature for up to 10,000 hours. Therefore, there is no risk of a significant effect on the mechanical properties of the confinement or boundary 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 that are not included in Table 3.3.1 are weight density and Poisson's ratio. These properties are assumed constant for all structural analyses, regardless of temperature. The values used are shown in the table below.

PROPERTY	VALUE
Weight Density (lb/in <sup>3</sup> )	0.290
Poisson's Ratio	0.30

b. Metamic-HT

Metamic-HT is a composite of nano-particles of aluminum oxide (alumina) and finely ground boron carbide particles dispersed in the metal matrix of pure aluminum. Metamic-HT is the principal constituent material of the HI-STORM FW fuel baskets. Metamic-HT neutron absorber is an enhanced version of the Metamic (classic) product widely used in dry storage fuel baskets [3.1.4, 3.3.2] and spent fuel storage racks [1.2.11]. The enhanced properties of Metamic-HT derive from the strengthening of its aluminum matrix with ultra fine-grained (nano-particle size) alumina (Al<sub>2</sub>O<sub>3</sub>) particles that anchor the grain boundaries. The strength properties of Metamic-HT have been characterized through a comprehensive test program, and Minimum Guaranteed Values suitable for structural design are archived in [Table 1.2.8]. The Metamic-HT metal matrix composite thus exhibits excellent mechanical strength properties (notably creep resistance) in addition to the proven thermal and neutron absorption properties that are intrinsic to borated aluminum materials. The specific Metamic-HT composition utilized in this FSAR has 10% (min.) B<sub>4</sub>C by weight.

Section 1.2.1.4.1 provides detailed information on Metamic-HT. Mechanical properties are provided in Table 1.2.8

c. Carbon Steel, Low-Alloy and Nickel Alloy Steel

The carbon steels in the HI-STORM FW system are SA516 Grade 70, SA515 Grade 70, and SA36. The low alloy steel is SA350-LF3. The material properties of SA516 Grade 70 and SA515 Grade 70 are shown in Tables 3.3.2. The material properties of SA350-LF2 and SA350-LF3 are given in Table 3.3.3. The material properties of SA36 are shown in Table 3.3.6.

Two properties of these steels that are not included in Tables 3.3.2, 3.3.3 and 3.3.6 are weight density and Poisson's ratio. These properties are assumed constant for all structural analyses. The values used are shown in the table below.

PROPERTY	VALUE
Weight Density (lb/in <sup>3</sup> )	0.283
Poisson's Ratio	0.30

d. Bolting Materials

Material properties of the bolting materials used in the HI-STORM FW system are given in Table 3.3.4.

e. Weld Material

All weld materials utilized in the welding of the Code components comply with the provisions of the appropriate ASME subsection (e.g., Subsection NB for the MPC enclosure vessel) and Section IX. All non-code welds will be made using weld procedures that meet Section IX of the ASME Code. 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

a. Concrete

The primary function of the plain concrete in the HI-STORM FW storage overpack is shielding. Concrete in the HI-STORM FW overpack is not considered as a structural member, except to withstand compressive, bearing, and penetrant loads. Therefore the mechanical behavior of concrete must be quantified to determine the stresses in the structural members (steel shells surrounding it) under accident conditions. Table 3.3.5 provides the concrete mechanical properties. Allowable, bearing strength in concrete for normal loading conditions is calculated in accordance with ACI 318-05 [3.3.5]. The procedure specified in ASTM C-39 is utilized to verify that the assumed compressive strength will be realized in the actual in-situ pours. Appendix 1.D in the HI-STORM 100 FSAR [3.1.4] provides additional information on the requirements on plain concrete for use in HI-STORM FW storage overpack.

To enhance the shielding performance of the HI-STORM FW storage overpack, high density concrete can be used during fabrication. The permissible range of concrete densities is specified in Table 1.2.5. The structural calculations consider the most conservative density value (i.e., maximum or minimum weight), as appropriate.



b. Lead

Lead is not considered as a structural member of the HI-STORM FW system. Its load carrying capacity is neglected in all structural analysis, except in the analysis of a tornado missile strike where it acts as a missile barrier. Applicable mechanical properties of lead are provided in Table 3.3.5.

c. Fuel Basket Shims

The fuel basket shims (basket shims), as presented on the drawings in Section 1.5, are made of an aluminum alloy to ensure a high thermal conductivity and to ensure stable mechanical properties in the temperature range obtained in the peripheral region of the fuel basket. Nominal mechanical properties for the basket shims are tabulated in Table 3.3.7.

Strictly speaking, the shim is not a structural material because it does not withstand any tensile loads and is located in a confined space which would prevent its uncontrolled deformation under load. The simulation of the shim in the basket's structural model, however, utilizes its mechanical properties of which only the Yield Strength has a meaningful (but secondary) role. Accordingly, in this FSAR, the nominal value of the Yield Strength specified in Table 3.3.7 herein, is set down as a "critical characteristic" for the shim material. The minimum value of the Yield Strength reported in the material supplier's CoC must be at least 90% of the nominal value in the above referenced table to ensure that the non-mechanistic tip-over analysis will not have to be revisited.

Table 3.3.1

## ALLOY X MATERIAL PROPERTIES

Temp. (Deg. F)	Alloy X			
	S <sub>y</sub>	S <sub>u</sub> <sup>†</sup>	α	E
-40	30.0	75.0 (70.0)	--	28.88
100	30.0	75.0 (70.0)	8.6	28.12
150	27.5	73.0 (68.1)	8.8	27.81
200	25.0	71.0 (66.3)	8.9	27.5
250	23.7	68.6 (64.05)	9.1	27.25
300	22.4	66.2 (61.8)	9.2	27.0
350	21.55	65.3 (60.75)	9.4	26.7
400	20.7	64.4 (59.7)	9.5	26.4
450	20.05	63.9 (59.45)	9.6	26.15
500	19.4	63.4 (59.2)	9.7	25.9
550	18.85	63.35 (59.1)	9.8	25.6
600	18.3	63.3 (59.0)	9.8	25.3
650	17.8	62.85 (58.6)	9.9	25.05
700	17.3	62.4 (58.3)	10.0	24.8
750	16.9	62.1 (57.9)	10.0	24.45
800	16.5	61.7 (57.6)	10.1	24.1

## Definitions:

S<sub>y</sub> = Yield Stress (ksi)α = Mean Coefficient of thermal expansion (in./in. per degree F x 10<sup>-6</sup>)S<sub>u</sub> = Ultimate Stress (ksi)E = Young's Modulus (psi x 10<sup>6</sup>)

## Notes:

1. Source for S<sub>y</sub> values is Table Y-1 of [3.3.1].
2. Source for S<sub>u</sub> values is Table U of [3.3.1].
3. Source for α 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].

<sup>†</sup> The ultimate stress of Alloy X is dependent on the product form of the material (i.e., forging vs. plate). Values in parentheses 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.

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Table 3.3.2

## SA516 AND SA515, GRADE 70 MATERIAL PROPERTIES

Temp. (Deg. F)	SA516 and SA515, Grade 70			
	S <sub>y</sub>	S <sub>u</sub>	$\alpha$	E
-40	38.0	70.0	---	29.98
100	38.0	70.0	6.5	29.26
150	35.7	70.0	6.6	29.03
200	34.8	70.0	6.7	28.8
250	34.2	70.0	6.8	28.55
300	33.6	70.0	6.9	28.3
350	33.05	70.0	7.0	28.1
400	32.5	70.0	7.1	27.9
450	31.75	70.0	7.2	27.6
500	31.0	70.0	7.3	27.3
550	30.05	70.0	7.3	26.9
600	29.1	70.0	7.4	26.5
650	28.2	70.0	7.5	26.0
700	27.2	70.0	7.6	25.5
750	26.3	69.1	7.7	24.85

## Definitions:

S<sub>y</sub> = Yield Stress (ksi) $\alpha$  = Mean Coefficient of thermal expansion (in./in. per degree F x 10<sup>-6</sup>)S<sub>u</sub> = Ultimate Stress (ksi)E = Young's Modulus (psi x 10<sup>6</sup>)

## Notes:

1. Source for S<sub>y</sub> values is Table Y-1 of [3.3.1].
2. Source for S<sub>u</sub> values is Table U of [3.3.1].
3. Source for  $\alpha$  values is material group 1 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]

Table 3.3.3

## SA350-LF3 AND SA350-LF2 MATERIAL PROPERTIES

Temp. (Deg. F)	SA350-LF3 (SA350-LF2)			SA350-LF3 (SA350-LF2)	
	S <sub>m</sub>	S <sub>y</sub>	S <sub>u</sub>	E	α
-20	23.3	37.5 (36.0)	70.0	28.22 (29.88)	---
100	23.3	37.5 (36.0)	70.0	27.64 (29.26)	6.5
200	22.9 (22.0)	34.3 (33.0)	70.0 (70.0)	27.1 (28.8)	6.7
300	22.1 (21.2)	33.2 (31.8)	70.0 (70.0)	26.7 (28.3)	6.9
400	21.4 (20.5)	32.0 (30.8)	70.0 (70.0)	26.2 (27.9)	7.1
500	20.3 (19.6)	30.4 (29.3)	70.0 (70.0)	25.7 (27.3)	7.3
600	18.8 (18.4)	28.2 (27.6)	70.0 (70.0)	25.1 (26.5)	7.4
700	16.9 (17.2)	25.3 (25.8)	66.5 (70.0)	24.6 (25.5)	7.6

## Definitions:

- S<sub>m</sub> = Design Stress Intensity (ksi)  
 S<sub>y</sub> = Yield Stress (ksi)  
 S<sub>u</sub> = Ultimate Stress (ksi)  
 α = Mean Coefficient of Thermal Expansion (in./in. per degree F x 10<sup>-6</sup>)  
 E = Young's Modulus (psi x 10<sup>6</sup>)

## Notes:

1. Source for S<sub>m</sub> values is Table 2A of [3.3.1].
2. Source for S<sub>y</sub> values is Table Y-1 of [3.3.1].
3. Source for S<sub>u</sub> values is ratioing S<sub>m</sub> values.
4. Source for α values is group 1 alloys in Table TE-1 of [3.3.1].
5. Source for E values is material group B (for SA350-LF3) and "Carbon steels with C less than or equal to 0.30%" (for SA350-LF2) in Table TM-1 of [3.3.1].
6. Values for LF2 are given in parentheses where different from LF3.

Table 3.3.4

## BOLTING MATERIAL PROPERTIES

SB637-N07718 (less than or equal to 6 inches diameter)					
Temp. (Deg. F)	S <sub>y</sub>	S <sub>u</sub>	E	$\alpha$	S <sub>m</sub>
-100	150.0	185.0	29.9	---	50.0
-20	150.0	185.0	29.43	---	50.0
70	150.0	185.0	28.9	7.1	50.0
100	150.0	185.0	28.76	7.1	50.0
200	144.0	177.6	28.3	7.2	48.0
300	140.7	173.5	27.9	7.3	46.9
400	138.3	170.6	27.5	7.5	46.1
500	136.8	168.7	27.2	7.6	45.6
600	135.3	166.9	26.8	7.7	45.1
SA193 Grade B7 (2.5 to 4 inches diameter)					
Temp. (Deg. F)	S <sub>y</sub>	S <sub>u</sub>	E	$\alpha$	S <sub>m</sub>
100	95.0	115.0	29.46	6.5	31.7
200	88.5	115.0	29.0	6.7	29.5
300	85.1	115.0	28.5	6.9	28.4
400	82.7	115.0	28.0	7.1	27.6
500	80.1	115.0	27.4	7.3	26.7
600	77.1	115.0	26.9	7.4	25.7

## Definitions:

S<sub>m</sub> = Design stress intensity (ksi)S<sub>y</sub> = Yield Stress (ksi) $\alpha$  = Mean Coefficient of thermal expansion (in./in. per degree F x 10<sup>-6</sup>)S<sub>u</sub> = Ultimate Stress (ksi)E = Young's Modulus (psi x 10<sup>6</sup>)

## Notes:

1. Source for S<sub>m</sub> values is Table 4 of [3.3.1].
2. Source for S<sub>y</sub> values is ratioing design stress intensity values and Table Y-1 of [3.3.1], as applicable.
3. Source for S<sub>u</sub> values is ratioing design stress intensity values and Table U of [3.3.1], as applicable.
4. Source for  $\alpha$  values is Tables TE-1 and TE-4 of [3.3.1], as applicable.
5. Source for E values is Tables TM-1 and TM-4 of [3.3.1], as applicable.

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Table 3.3.4 (CONTINUED)

## BOLTING MATERIAL PROPERTIES

Temp. (Deg. F)	$S_y$	$S_u$	E	$\alpha$	$S_m$
SA193 Grade B7 (less than or equal to 2.5 inches diameter)					
100	105.0	125.0	29.46	6.5	35.0
200	98.0	125.0	29.0	6.7	32.6
300	94.1	125.0	28.5	6.9	31.4
400	91.5	125.0	28.0	7.1	30.5
500	88.5	125.0	27.4	7.3	29.5
600	85.3	125.0	26.9	7.4	28.4
SA564 630 H1150					
100	105.0	135.0	28.33	6.4	45.0
200	97.1	135.0	27.8	6.6	45.0
300	93.0	135.0	27.2	6.7	45.0
400	89.7	131.2	26.7	6.9	43.7
500	87.0	128.6	26.1	7.0	42.9
600	84.7	126.7	25.5	7.1	42.2

## Definitions:

 $S_m$  = Design stress intensity (ksi) $S_y$  = Yield Stress (ksi) $\alpha$  = Mean Coefficient of thermal expansion (in./in. per degree F x  $10^{-6}$ ) $S_u$  = Ultimate Stress (ksi)E = Young's Modulus (psi x  $10^6$ )

## 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  $\alpha$  values is group 1 alloys and Condition 1150 precipitation hardened stainless steel in Table TE-1 of [3.3.1].
4. Source for E values is material group C and S17400 in Table TM-1 of [3.3.1].
5. Source for  $S_m$  values is Table 4 and Table 2A of [3.3.1] for SA-193 Grade B7 and SA-564 630 H1150 respectively.

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Table 3.3.5						
CONCRETE AND LEAD MECHANICAL PROPERTIES						
PROPERTY	VALUE					
CONCRETE:						
Compressive Strength (psi)	3,300 psi					
Nominal Density (lb/ft³)	150 lb/cubic feet					
Allowable Bearing Stress (psi)	1,543 <sup>†</sup>					
Allowable Axial Compression (psi)	1,042 <sup>†</sup>					
Allowable Flexure, extreme fiber tension (psi)	158 <sup>†,††</sup>					
Allowable Flexure, extreme fiber compression (psi)	1,543 <sup>†</sup>					
Mean Coefficient of Thermal Expansion (in/in/deg. F)	5.5E-06					
Modulus of Elasticity (psi)	57,000 (compressive strength (psi)) <sup>1/2</sup>					
LEAD:	-40°F	-20°F	70°F	200°F	300°F	600°F
Yield Strength (psi)	700	680	640	490	380	20
Modulus of Elasticity (ksi)	2.4E+3	2.4E+3	2.3E+3	2.0E+3	1.9E+3	1.5E+3
Coefficient of Thermal Expansion (in/in/deg. F)	15.6E-6	15.7E-6	16.1E-6	16.6E-6	17.2E-6	20.2E-6
Poisson's Ratio	0.40					
Density (lb/cubic ft.)	708					

Notes:

- Concrete allowable stress values based on ACI 318-05.
- Lead properties are from [3.3.7].

<sup>†</sup> Values listed correspond to concrete compressive stress = 3,300 psi.

<sup>††</sup> No credit for tensile strength of concrete is taken in the calculations.

Table 3.3.6				
SA36 MATERIAL PROPERTIES				
Temp. (Deg. F)	SA36			
	S <sub>y</sub>	S <sub>u</sub>	α	E
-40	36.0	58.0	---	29.98
100	36.0	58.0	6.5	29.26
150	33.8	58.0	6.6	29.03
200	33.0	58.0	6.7	28.8
250	32.4	58.0	6.8	28.55
300	31.8	58.0	6.9	28.3
350	31.3	58.0	7.0	28.1
400	30.8	58.0	7.1	27.9
450	30.05	58.0	7.2	27.6
500	29.3	58.0	7.3	27.3
550	28.45	58.0	7.3	26.9
600	27.6	58.0	7.4	26.5
650	26.7	58.0	7.5	26.0
<sup>+</sup> 700	25.8	58.0	7.6	25.5

## Definitions:

S<sub>y</sub> = Yield Stress (ksi)α = Mean Coefficient of thermal expansion (in./in./°F x 10<sup>-6</sup>)S<sub>u</sub> = Ultimate Stress (ksi)E = Young's Modulus (psi x 10<sup>6</sup>)

## Notes:

1. Source for S<sub>y</sub> values is Table Y-1 of [3.3.1].
2. Source for S<sub>u</sub> values is Table U of [3.3.1].
3. Source for α values is group 1 alloys 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].



Table 3.3.7					
FUEL BASKET SHIMS – NOMINAL MECHANICAL PROPERTIES					
Temp. °C (°F)	Aluminum Alloy (B221 2219-T8511)				
	S <sub>y</sub>	S <sub>u</sub>	E	α	% Elongation
25 (75)	290 (42)	400 (58)	7.2 (10.5)	–	5
150 (300)	243 (35)	307 (44)	6.8 (9.8)	23.9 (13.3)	6.4
204 (400)	188 (27)	231 (34)	6.3 (9.1)	24.5 (13.6)	8.2
230 (450)	171 (25)	209 (30)	6.1 (8.8)	24.8 (13.8)	8.6
260 (500)	154 (22)	182 (26)	5.9 (8.5)	25.0 (13.9)	8.6
290 (550)	98 (14)	116 (17)	5.5 (8.0)	25.4 (14.1)	10.5

## Definitions:

S<sub>y</sub> = Yield Stress, MPa (ksi)α = Mean Coefficient of thermal expansion, cm/cm-°C x 10<sup>-6</sup> (in/in-°F x 10<sup>-6</sup>)S<sub>u</sub> = Ultimate Stress, MPa (ksi)E = Young's Modulus, MPa x 10<sup>4</sup> (psi x 10<sup>6</sup>)

## Notes:

1. Source for E values is “Properties of Aluminum Alloys”, page 82 [3.3.3] (properties listed in the table above are not affected by time at temperature).
2. Source for S<sub>y</sub>, S<sub>u</sub>, and % Elongation values at room temperature is ASTM Specification B221M [3.3.8]. Values at elevated temperatures are obtained by scaling the room temperature values using the data from [3.3.3].
3. Source for α is Table TE-2 of [3.3.1] (values listed in TE-2 are also considered representative of Aluminum Alloy (2219-T8511) (UNS No. A92219)).
4. The optional solid shims have no mechanical property requirements.

## 3.4 GENERAL STANDARDS FOR CASKS

### 3.4.1 Chemical and Galvanic Reactions

Chapter 8 provides discussions on chemical and galvanic reactions, material compatibility and operating environments. Section 8.12 provides a summary of compatibility all HI-STORM FW system materials with the operating environment.

### 3.4.2 Positive Closure

There are no quick-connect/disconnect ports in the Confinement Boundary of the HI-STORM FW system. The only access to the MPC is through the storage overpack lid, which weighs over 10 tons (see Table 3.2.5). The lid is fastened to the storage overpack with large bolts. Inadvertent opening of the storage overpack is not feasible because opening a storage overpack requires mobilization of special tools and heavy-load lifting equipment.

### 3.4.3 Lifting Devices

#### 3.4.3.1 Identification of Lifting Devices and Required Safety Factors

The safety of the lifting and handling operations involving HI-STORM FW system components is considered in this section. In particular, the compliance of the appurtenances integral to the cask components used in the lifting operations to NUREG-0612, Reg. Guide 3.61, and the ASME Code is evaluated.

The following design features of Threaded Anchor Locations (TALs) are relevant to their stress analysis:

- i. All TALs consist of vertically tapped penetrations in the solid metal blocks. For example, the HI-STORM FW overpack body and overpack lid (like all HI-STORM models) have tapped holes in the “anchor blocks” that are engaged for lifting. The loaded MPC is lifted at eight threaded penetrations in the top lid as depicted on the licensing drawings in Section 1.5. However, the MPC lifting analysis in this section conservatively takes credit for only 4 TALs. Likewise, eight vertically tapped holes in the top flange provide the lift points for HI-TRAC VW transfer cask.

Specifically, trunnions are not used in the HI-STORM FW system components because of the radiation streaming paths introduced by their presence and high stresses produced at the trunnion’s root by the cantilever action during lifting.

- ii. Operations involving loaded HI-STORM FW cask components involve handling evolutions in the vertical orientation (with the rare handling exception of the transfer cask as described in Subsection 4.5.1). While the lifting devices used by a specific nuclear site shall be custom engineered to meet the architectural constraints of the site, all lifting devices are required to

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engage the tapped connection points using a vertical tension member such as a threaded rod. Thus, the loading on the cask during lifting is purely vertical.

- iii. There are no rotation trunnions in the HI-STORM FW components. All components are upended and downended at the nuclear plant site using “cradles” of the same design used at the factory (viz., the Holtec Manufacturing Division) during their manufacturing.

The stress analysis of the HI-STORM FW components, therefore, involves applying a vertical load equal to  $D^*/n$  at each of the  $n$  TAL locations. Thus, for the case of the HI-STORM FW overpack,  $n = 4$  (four “anchor blocks” as shown in the licensing drawings in Section 1.5).

The stress limits for individual components are as follows:

- i. Lift points (MPC and HI-TRAC VW): The stress in the threads must be the lesser of  $1/3^{\text{rd}}$  of the material’s yield strength and  $1/10^{\text{th}}$  of its ultimate strength pursuant to NUREG-0612 and Reg. Guide 3.61.
- ii. Lift points (HI-STORM FW): The stress in the threads must be less than  $1/3^{\text{rd}}$  of the material’s yield strength pursuant to Reg. Guide 3.61. This acceptance criterion is consistent with the stress limits used for the lifting evaluation of the HI-STORM 100 overpack in [3.1.4].
- iii. Balance of the components: The maximum primary stress (membrane plus bending) must be below the Level A service condition limit using ASME Code, Section III, Subsection NF (2007 issue) as the reference code.

To incorporate an additional margin of safety in the reported safety factors, the following assumptions are made:

- i. As the system description in Chapter 1 indicates, the heights of the MPCs, HI-STORM FW and HI-TRAC VW are variable. Further, the quantity of lead shielding installed in HI-TRAC VW and the density of concrete can be increased to maximize shielding. All lift point capacity evaluations are performed using the maximum possible weights for each component, henceforth referred to as the “heaviest weight configuration”. Because a great majority of site applications will utilize lower weight components (due to shorter fuel length and other architectural limitations such as restricted crane capacity or DAS slab load bearing capacity, or lack of floor space in the loading pit), there will be an additional margin of safety in the lifting point’s capacity at specific plant sites.
- ii. All material yield strength and ultimate strength values used are the minimum from the ASME Code. Actual yield and tensile data for manufactured steel usually have up to 20% higher values.

The stress analysis of the lifting operation is carried out using the load combination  $D+H$ , where  $H$  is the “handling load”. 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. 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 FSAR use the "apparent dead load",  $D^*$ , as the lifted load.

Unless explicitly stated otherwise, all analyses of lifting operations presented in this FSAR 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

$$\text{Safety Factor, } \beta = \frac{\text{Allowable Stress}}{\text{Computed Stress}}$$

In the following subsections, the lifting device stress analyses performed to demonstrate compliance with regulations are presented. Summary results are presented for each of the analyses.

### 3.4.3.2 Analysis of Lifting Scenarios

In the following, the safety analyses of the HI-STORM FW components under the following lifting conditions are summarized.

#### a. MPC Lifts

The governing condition for the MPC lift is when it is being raised or lowered in a radiation shielded space defined by the HI-TRAC VW or HI-STORM FW stack. In this condition, as stated in Section 3.4.3.1, only four tapped holes in the MPC lid (Alloy X material) are credited to carry the weight.

The criteria derived from NUREG-0612, Reg. Guide 3.61, and the ASME Code Level A condition, stated earlier, apply. The stress analysis is carried out in two parts.

- i. Strength analysis of the TALs (connection points) using classical strength-of-materials.
- ii. A finite element analysis of the MPC as a cylindrical vessel with the weight of the fuel and basket applied on its baseplate which along with the weight of the Confinement Boundary metal is equilibrated by the reaction loads at the four lift points.

The primary stress intensities must meet the Level A stress limits for "NB" Class 3 plate and shell structures.

#### Case (i): Stress Analysis of MPC Threaded Anchor Locations (TALs)

Per Table 3.2.8, the maximum weight of a loaded MPC is

$$D = 116,400 \text{ lb}$$

Per the above, the apparent dead load of the MPC during handling operations is

$$D^* = 1.15 \times D = 133,860 \text{ lb}$$

The MPC lid has 8 TALs as shown on the drawings in Section 1.5, but as stated in Section 3.4.3.1, only four tapped holes in the MPC lid are credited to carry the weight. Therefore, the lifted load per TAL is equal to

$$\frac{D^*}{4} = 33,465 \text{ lb}$$

Per Machinery's Handbook [3.4.12], the shear area of the internal threads (1 3/4" - 5UNC x 3.0" min length.) at each TAL is computed as

$$A = 11.8 \text{ in}^2$$

Finally, the shear stress on the TALs is computed as follows

$$\tau = \frac{D^*}{4A} = 2,836 \text{ psi}$$

The MPC lid is made from Alloy X material, whose mechanical properties are listed in Table 3.3.1. Based on a design temperature of 600°F (Table 2.2.3), and assuming the yield and ultimate strengths in shear to be 60% of the corresponding tensile strengths, the allowable stress in the threads is determined as follows

$$Sa = 0.6 \times \min\left(\frac{Sy}{3}, \frac{Su}{10}\right) = 3,540 \text{ psi}$$

Therefore, the safety factor against shear failure of the TALs in the MPC lid is

$$SF = \frac{Sa}{\tau} = 1.248$$

#### Case (ii): Finite Element Analysis of MPC Enclosure Vessel

The stress analysis of the MPC Enclosure Vessel under normal handling conditions is performed using ANSYS [3.4.1]. The finite element model, which is shown in Figure 3.4.1, is 1/4 -symmetric,

and it represents the maximum height MPC as defined by Tables 3.2.1 and 3.2.2. The maximum height MPC is analyzed because it is also the heaviest MPC. The key attributes of the ANSYS finite element model of the MPC Enclosure Vessel are described in Subsection 3.1.3.2.

The loads are statically applied to the finite element model in the following manner. The self weight of the Enclosure Vessel is simulated by applying a constant acceleration of 1.15g in the vertical direction. The apparent dead weight of the stored fuel inside the MPC cavity (which includes a 15% dynamic amplifier) is accounted for by applying a uniformly distributed pressure of 18.8 psi on the top surface of the MPC baseplate. The amplified weight of the fuel basket and the fuel basket shims is applied as a ring load on the MPC baseplate at a radius equal to the half-width of the fuel basket cross section. The magnitude of the ring load is equal to 100.4 lbf/in. All internal surfaces of the MPC storage cavity are also subjected to an internal pressure of 95 psig, which exceeds the normal operating pressure per Table 4.4.5. Finally, the model is constrained by fixing one node on the top surface of the ¼-symmetric MPC lid, which coincides with the TAL. Symmetric boundary conditions are applied to the two vertical symmetry planes. The boundary conditions and the applied loads are graphically depicted in Figure 3.4.28.

The resulting stress intensity distribution in the Enclosure Vessel under the applied handling loads is shown in Figure 3.4.2. Figures 3.4.29 and 3.4.30 plot the thru-thickness variation of the stress intensity at the baseplate center and at the baseplate-to-shell juncture, respectively. The maximum primary and secondary stress intensities in the MPC Enclosure Vessel are compared with the applicable stress intensity limits from Subsection NB of the ASME Code [3.4.4]. The allowable stress intensities are taken at 450°F for the MPC shell and MPC lids, 300°F for the baseplate, and 250°F at the baseplate-to-shell juncture. These temperatures bound the operating temperatures for these parts under normal operating conditions (Table 4.4.3). The maximum calculated stress intensities and the corresponding safety factors are summarized in Table 3.4.1.

The shear stress in the MPC lid-to-shell weld under normal handling conditions is independently calculated, as shown below.

Per Table 3.2.8, the maximum weight of a loaded MPC is

$$W_{MPC} = 116,400 \text{ lb}$$

The diameter and weight of the MPC lid assembly are

$$D = 74.5 \text{ in}$$

$$W_{lid} = 11,500 \text{ lb}$$

From Table 4.4.5, the bounding pressure inside the MPC cavity under normal operating conditions is

$$P = 95 \text{ psig}$$

Thus, the total force acting on the MPC lid-to-shell weld is

$$F = 1.15 \cdot (W_{MPC} - W_{lid}) + P \cdot \left( \frac{\pi \cdot D^2}{4} \right) = 534,755 lb$$

which includes a 15% dynamic amplifier. The MPC lid-to-shell weld is a 3/4" partial groove weld, which has an effective area equal to

$$A = \pi \cdot D \cdot \left( t_w - \frac{1}{8} \text{ in} \right) \cdot 0.8 = 117.0 \text{ in}^2$$

where  $t_w$  is the weld size (= 0.75 in). The calculated weld area includes a strength reduction factor of 0.8 per ISG-15 [3.4.17]. Thus, the average shear stress in the MPC lid-to-shell weld is

$$\tau = \frac{F}{A} = 4,571 \text{ psi}$$

The MPC Enclosure Vessel is made from Alloy X material, whose mechanical properties are listed in Table 3.3.1. Based on a temperature of 450°F (Table 4.4.3), and assuming that the weld strength is equal to the base metal ultimate strength, the allowable shear stress in the weld under normal conditions is

$$\tau_a = 0.3 \times S_u = 19,170 \text{ psi}$$

Therefore, the safety factor against shear failure of the MPC lid-to-shell weld is

$$SF = \frac{\tau_a}{\tau} = 4.19$$

#### b. Heaviest Weight HI-TRAC VW Lift

The HI-TRAC VW transfer cask is at its heaviest weight when it is being lifted out of the loading pit with the MPC full of fuel and water and the MPC lid lying on it for shielding protection (Table 3.2.8). The threaded lift points provide for the anchor locations for lifting.

The stress analysis of the transfer cask consists of two steps:

- i. A strength evaluation of the tapped connection points to ensure that it will not undergo yielding at 3 times  $D^*$  and failure at 10 times  $D^*$ .
- ii. A strength evaluation of the HI-TRAC VW vessel using strength of materials formula to establish the stress field under  $D^*$ . The primary membrane plus primary bending stresses

throughout the HI-TRAC VW body and the bottom lid shall be below the Level A stress limits for “NF” Class 3 plate and shell structures.

Case (i): Stress Analysis of HI-TRAC VW Threaded Anchor Locations (TALs)

Per Table 3.2.8, the maximum lifted weight of a loaded HI-TRAC VW is

$$D = 270,000 \text{ lb}$$

Per the above, the apparent dead load of the HI-TRAC VW during handling operations is

$$D^* = 1.15 \times D = 310,500 \text{ lb}$$

The HI-TRAC VW top flange has 8 TALs as shown on the drawing in Section 1.5. Therefore, the lifted load per TAL is equal to

$$\frac{D^*}{8} = 38,813 \text{ lb}$$

Per Machinery’s Handbook [3.4.12], the shear area of the internal threads (2 1/4” - 4.5UNC x 2.25” min length.) at each TAL is

$$A = 12.228 \text{ in}^2$$

Finally, the shear stress on the TALs is computed as follows

$$\tau = \frac{D^*}{8A} = 3,174 \text{ psi}$$

The HI-TRAC VW top flange is made from SA-350 LF3 material, whose mechanical properties are listed in Table 3.3.3. Based on a design temperature of 400°F (Table 2.2.3), and assuming the yield and ultimate strengths in shear to be 60% of the corresponding tensile strengths, the allowable stress in the threads is determined as follows

$$Sa = 0.6 \times \min\left(\frac{Sy}{3}, \frac{Su}{10}\right) = 4,200 \text{ psi}$$

Therefore, the safety factor against shear failure of the TALs in the HI-TRAC VW top flange is

$$SF = \frac{Sa}{\tau} = 1.323$$



### Case (ii): Stress Analysis of HI-TRAC VW Body

The stress analysis of the HI-TRAC VW steel structure during lifting operations is performed using strength of materials. All structural members in the load path are evaluated for the maximum lifted weight (Table 3.2.8). In particular, the following stresses are calculated:

- the shear stress in the welds between the top flange and the inner and outer shells
- the primary membrane stress in the inner and outer shells
- the tensile stress in the bottom lid bolts
- the primary bending stress in the bottom lid

To determine the bending stress in the bottom lid, the weight of the loaded MPC (Table 3.2.8) plus the weight of the water inside the HI-TRAC VW cavity (Table 3.2.4) is applied as a uniformly distributed pressure on the top surface of the lid. The bending stress is calculated at the center of the bottom lid assuming that the lid is simply supported at the bolt circle diameter. The calculated stresses are compared with the Level A stress limits for “NF” Class 3 plate and shell structures. The detailed calculations are documented in [3.4.13]. Table 3.4.2 summarizes the stress analysis results for the HI-TRAC VW steel structure under the maximum lifted load.

### c. HI-STORM FW Overpack Related Lifts

Two related lift conditions are:

- i. HI-STORM FW loaded with the heaviest MPC and closure lid installed being lifted (heaviest weight configuration).
- ii. HI-STORM FW lid being lifted (heaviest weight configuration)

### Case (i): HI-STORM FW Lift Using Anchor Block Connections

Calculations to establish the margin of safety in the TALs and the HI-STORM FW overpack’s steel structure are summarized below.

Per Table 3.2.8, the maximum weight of a loaded HI-STORM FW is

$$D = 425,700 \text{ lb}$$

Per the above, the apparent dead load of the HI-STORM FW during handling operations is

$$D^* = 1.15 \times D = 489,555 \text{ lb}$$

The HI-STORM FW overpack has 4 TALs as shown on the drawing in Section 1.5. Therefore, the lifted load per TAL is equal to

$$\frac{D^*}{4} = 122,389/lb$$

Per Machinery's Handbook [3.4.12], the shear area of the internal threads (3 1/4" - 4UNC x 3 1/4" min length.) at each TAL is

$$A = 24.1 \text{ in}^2$$

Finally, the shear stress on the TALs is computed as follows

$$\tau = \frac{D^*}{4A} = 5,072 \text{ psi}$$

The HI-STORM FW anchor blocks are made from SA-350 LF2 material, whose mechanical properties are listed in Table 3.3.3. Based on a design temperature of 350°F (Table 2.2.3), and assuming the yield strength in shear to be 60% of the corresponding tensile yield strength, the allowable stress in the threads is determined as follows

$$Sa = 0.6 \times \frac{Sy}{3} = 6,260 \text{ psi}$$

Therefore, the safety factor against shear failure of the TALs in the HI-STORM FW overpack is

$$SF = \frac{Sa}{\tau} = 1.234$$

The stress analysis of the overpack body under normal handling conditions is performed using ANSYS [3.4.1]. The finite element model, which is shown in Figure 3.4.3, is 1/4-symmetric, and it represents the maximum height HI-STORM FW as defined by Tables 3.2.1 and 3.2.2. The concrete density is also maximized (Table 3.2.5) in the ANSYS model. The key attributes of the ANSYS finite element model of the HI-STORM FW overpack are described in Subsection 3.1.3.1.

The self weight of the overpack is simulated by applying a constant acceleration of 1.15g in the vertical direction. The apparent dead weight of the fully loaded MPC (which includes a 15% dynamic amplifier) is accounted for by applying a uniformly distributed pressure of 23.8 on the top surface of the HI-STORM FW baseplate. Finally, the model is constrained by fixing four nodes on the top surface of the HI-STORM FW, which coincide with the TALs. Symmetric boundary conditions are applied to the two vertical symmetry planes. The boundary conditions and the applied loads are graphically depicted in Figure 3.4.26.

The resulting stress distribution in the overpack under the applied handling loads is shown in Figure 3.4.4. The maximum primary stresses in the HI-STORM overpack body are compared with the applicable stress limits from Subsection NF of the ASME Code [3.4.2]. The allowable stresses for the load-bearing members are taken at 350°F, which exceeds the maximum operating temperature for the overpack under normal operating conditions (Table 4.4.3 and 4.4.14). The maximum stresses and the corresponding safety factors are summarized in Table 3.4.3.

#### Case (ii): Lid Lift Analysis

The weight of the HI-STORM FW lid is dependent on the shielding concrete's density. The maximum possible weight of the lid is provided in Table 3.2.5. The HI-STORM FW lid is lifted using the four equally spaced TALs on the lid top surface, which are shown on the licensing drawing in Section 1.5. Calculations to establish the margin of safety in the TALs and the lid's steel structure are summarized below.

Per Table 3.2.5, the maximum weight of the HI-STORM FW lid is

$$D = 23,300 \text{ lb}$$

Per the above, the apparent dead load of the HI-STORM FW lid during handling operations is

$$D^* = 1.15 \times D = 26,795 \text{ lb}$$

The HI-STORM FW lid has 4 TALs as shown on the drawing in Section 1.5. Therefore, the lifted load per TAL is equal to

$$\frac{D^*}{4} = 6,699 \text{ lb}$$

Per Machinery's Handbook [3.4.12], the shear area of the internal threads (1 1/2" - 6UNC x 1" min length.) at each TAL is

$$A = 3.567 \text{ in}^2$$

Finally, the shear stress on the TALs is computed as follows

$$\tau = \frac{D^*}{4A} = 1878 \text{ psi}$$

The HI-STORM FW lid anchor blocks are made from carbon steel material, whose yield and ultimate strengths at 450°F (Table 2.2.3) are conservatively input as 15,000 psi and 40,000 psi, respectively. Assuming the yield and ultimate strengths in shear to be 60% of the corresponding tensile strengths, the allowable stress in the threads is determined as follows

$$S_a = 0.6 \times \min\left(\frac{S_y}{3}, \frac{S_u}{10}\right) = 2,400 \text{ psi}$$

Therefore, the safety factor against shear failure of the TALs in the HI-STORM FW lid is

$$SF = \frac{S_a}{\tau} = 1.278$$

The global stress analysis of the overpack lid under normal handling conditions is performed using ANSYS [3.4.1]. Figure 3.4.5 shows the finite element model of the lid, which incorporates the maximum concrete density (Table 3.2.5). The key attributes of the ANSYS finite element model of the HI-STORM FW lid are described in Subsection 3.1.3.1.

The self weight of the overpack lid is simulated by applying a constant acceleration of 1.15g in the vertical direction. The model is constrained by fixing four nodes on the top surface of the HI-STORM FW lid, which coincide with the TALs.

The resulting stress distribution in the steel structure of the overpack lid under the applied handling load is shown in Figure 3.4.6. The maximum stresses and the corresponding safety factors are summarized in Table 3.4.4. For conservatism, the maximum calculated stress at any point on the lid, including secondary stress contributions, is compared against the primary membrane and primary bending stress limits per Subsection NF of the ASME Code for Level A conditions. The allowable stresses are taken at 300°F, which exceeds the maximum operating temperature for the overpack top lid under normal operating conditions.

### 3.4.3.3 Safety Evaluation of Lifting Scenarios

As can be seen from the above, the computed factors of safety have a large margin over the allowable (of 1.0) in every case. In the actual fabricated hardware, the factors of safety will likely be much greater because of the fact that the actual material strength properties are generally substantially greater than the Code minimums. Minor variations in manufacturing, on the other hand, may result in a small subtraction from the above computed factors of safety. A part 72.48 safety evaluation will be required if the cumulative effect of manufacturing deviation and use of the CMTR (or CoC) material strength in a manufactured hardware renders a factor of safety to fall below the above computed value. Otherwise, a part 72.48 evaluation is not necessary. The above criterion applies to all lift calculations covered in this FSAR.

### 3.4.4 Heat

The thermal evaluation of the HI-STORM FW system is reported in Chapter 4.

#### a. 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.

#### b. Differential Thermal Expansion

The effect of differential thermal expansion among the constituent components in the HI-STORM FW system is considered in Chapter 4 wherein the temperatures necessary to perform the differential thermal expansion analyses for the MPC in the HI-STORM FW and HI-TRAC VW casks are computed. The material presented in Section 4.4 demonstrates that a constraint to free expansion due to differential growth between discrete components of the HI-STORM FW system (e.g., storage overpack and enclosure vessel) will not develop under any operating condition.

##### i. Normal Hot Environment

Results presented in Section 4.4 demonstrate that initial gaps between the HI-STORM FW storage overpack or the HI-TRAC VW transfer cask and the MPC canister, and between the MPC canister and the fuel basket, will not close due to thermal expansion of the system components normal operating conditions.

The clearances between the MPC basket and canister structure, as well as between the MPC shell and storage overpack or HI-TRAC VW inside surface, are shown in Section 4.4 to be sufficient to preclude a temperature induced interference from differential thermal expansions under normal operating conditions.

##### ii. Fire Accident

It is shown in Chapter 4 that the fire accident has a small effect on the MPC temperatures because of the short duration of the fire accidents and the large thermal inertia of the storage overpack. Therefore, a structural evaluation of the MPC under the postulated fire event is not required. The conclusions reached in item (i) above are also appropriate for the fire accident with the MPC housed in the storage overpack. Analysis of fire accident temperatures of the MPC housed within the HI-TRAC VW for thermal expansion is unnecessary, as the HI-TRAC VW, directly exposed to the fire, expands to increase the gap between the HI-TRAC VW and MPC.

As expected, the external surfaces of the HI-STORM FW storage overpack that are directly exposed to the fire event experience maximum rise in temperature. The outer shell and top plate in the top lid are the external surfaces that are in direct contact with heated air from fire. Table 4.6.2 provides the maximum temperatures attained at the key locations in HI-STORM FW storage overpack under the

postulated fire event.

The following conclusions are evident from the above table.

- The maximum metal temperature of the carbon steel shell most directly exposed to the combustion air is well below 700°F (Table 2.2.3 applicable short-term temperature limit). 700°F is the permissible temperature limit in the ASME Code for the outer shell material.
- The bulk temperature of concrete is well below the normal condition temperature limit of 300°F specified in Table 2.2.3. ACI-349-85 [3.3.6] permits 350°F as the short-term temperature limit; the shielding concrete in the HI-STORM FW overpack. As the detailed information in Section 4.6 shows, the radial extent in the concrete where the local temperature exceeds 350°F begins at the outer shell/concrete interface and ends in less than one-inch. Therefore, the potential loss in the shielding material's effectiveness is less than 4% of the concrete shielding mass in the overpack annulus.
- The metal temperature of the inner shell does not exceed 300°F at any location, which is well below the accident condition temperature specified in Table 2.2.3 for the inner shell.
- The presence of a vented space at the top of the overpack body ensures that there will be no pressure buildup in the concrete annulus due to the evaporation of vapor and gaseous matter from the shielding concrete.

Thus, it is concluded that the postulated fire event will not jeopardize the structural integrity of the HI-STORM FW overpack or significantly diminish its shielding effectiveness.

The above conclusions, as relevant, also apply to the HI-TRAC VW fire considered in Chapter 4. Water jacket over-pressurization is prevented by the pressure relief devices. The non-structural effects of loss of water have been evaluated in Chapter 5 and shown to meet regulatory limits. Therefore, it is concluded that the postulated fire event will not cause a state of non-compliance with the regulations to materialize.

#### **3.4.4.1 Safety Analysis**

Calculations of the stresses and displacements in the different components of the HI-STORM FW system from the effects of mechanical load case assembled in Table 3.1.1 for the MPC, the HI-STORM FW storage overpack and the HI-TRAC VW transfer cask are presented in the following. The purpose of the analyses summarized herein 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 of fuel from the MPC (for normal and off-normal conditions of storage) and the MPC from the HI-STORM FW storage overpack or from the HI-TRAC VW transfer cask.

Because many of the analyses must be performed for a particular ISFSI to demonstrate the acceptability of site-specific loads under the provisions of 10CFR72.212, the analyses presented here also set down the acceptable methodologies. Accordingly, the analysis methodologies are configured to exaggerate the severity of response. Also, because the weight and height of all three components (overpack, MPC, and HI-TRAC VW) can vary between specified ranges (see tables in Section 3.2), each analysis is carried out for the dimensional and weight condition of the component that maximizes response. Thus, for example, the seismic stability analysis of the loaded HI-STORM FW (Load Case 2 in Table 3.1.1) is performed for the case of maximum height, but the stability under the impact of a large tornado missile (Load Case 3) is analyzed assuming maximum height and minimum loaded weight (Per Table 3.2.8).

Each load case in Table 3.1.1 is considered sequentially and all affected components are analyzed to determine the factors of safety.

All factors of safety reported in this FSAR utilize nominal dimensions and minimum material strengths. Actual factors of safety in the manufactured hardware are apt to be considerably larger than those reported herein chiefly because of the actual material strengths being much greater than the values used in the safety analyses. A part 72.48 safety assessment will be required if the combined effect of the actual material strength and manufacturing deviation produces a lower safety factor for a design basis loading than that referenced in the safety evaluation in this FSAR.

#### 3.4.4.1.1 Load Case 1: Moving Floodwater

The object of the analysis is to determine the maximum floodwater velocity that a loaded HI-STORM FW on the ISFSI pad can withstand before tipping over or sliding. The flood data for the ISFSI shall be based on a 40-year (minimum) return flood. The kinematic stability analysis consists of writing static equilibrium equations for tipping and sliding.

The flood condition subjects the HI-STORM FW system to external pressure, together with a horizontal load due to water velocity. Because the HI-STORM FW storage overpack is equipped with ventilation openings, the hydrostatic pressure from flood submergence acts only on the MPC. As stated in Subsection 2.2.3, the design external pressure for the MPC bounds the hydrostatic pressure from flood submergence.

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.

The overturning horizontal force,  $F$ , due to hydraulic drag, is given by the classical formula:

$$F = C_d A V^* \quad [\text{Equation 1}]$$

where:

$$V^* = \text{velocity head} = \frac{\rho V^2}{2g} \quad (\rho \text{ is water weight density, and } g \text{ is acceleration due to gravity}).$$

$A$  = projected area of the HI-STORM FW cylinder perpendicular to the fluid velocity vector, equal to  $D$  times  $h$ , where  $h$  is the height of the floodwater.

$C_d$  = drag coefficient

The value of  $C_d$  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-STORM FW's sliding is opposed by the friction force, which is given by

$$F_f = \mu W^* \quad [\text{Equation 2}]$$

where:

$\mu$  = limiting value of the friction coefficient at the HI-STORM FW/ISFSI pad interface is assumed to be equal to 0.53 (the NRC-approved value in Docket No. 72-1014).

$W^*$  = apparent (buoyant) weight of HI-STORM FW with an empty MPC.

i. Sliding Factor of Safety

The factor of safety against sliding,  $\beta_1$ , is given by

$$\beta_1 = \frac{F_f}{F} = \frac{\mu W^*}{C_d A V^*} = \frac{2g\mu W^*}{C_d(Dh)\rho V^2} \quad [\text{Equation 3}]$$

The factor of safety,  $\beta_1$ , must be greater than 1.1. For  $g = 32.2 \text{ ft/sec}^2$ ,  $C_d = 0.5$ , and  $\rho = 62.4 \text{ lbf/ft}^3$ , the maximum value of  $V$  as a function of the floodwater height  $h$  is given by

$$V = \sqrt{\frac{1.876\mu W^*}{Dh}} \quad [\text{Equation 4}]$$



ii. Overturning Factor of Safety

For determining the margin of safety against overturning,  $\beta_2$ , the cask is assumed to pivot about a fixed point located at the outer edge of the contact circle at the interface between HI-STORM FW and the ISFSI. The overturning moment due to the hydraulic force  $F_T$  is balanced by a restoring moment from the buoyant weight acting at radius  $D/2$ .

Overturning moment,  $M_o = Fh/2$  where  $F$  is given by Equation 1 above.

Restoring moment,  $M_r = W^* D/2$  [Equation 5]

For stability against tipping  $M_o \leq M_r$

or  $Fh \leq W^* D$

Hence the factor of safety against overturning is

$$\beta_2 = \frac{W^* D}{Fh} = \frac{W^* D}{Cd A V^* h} = \frac{2gW^*}{Cdh^2 \rho V^2} \quad [\text{Equation 6}]$$

$\beta_2$  must be greater than 1.1. For  $g = 32.2 \text{ ft/sec}^2$ ,  $Cd = 0.5$ , and  $\rho = 62.4 \text{ lbf/ft}^3$ , the maximum value of  $V$  as a function of the floodwater height  $h$  is given by

$$V = \frac{\sqrt{1.876W^*}}{h} \quad [\text{Equation 7}]$$

The smaller of the value of  $V$  from Equations 4 and 7 defines the maximum permissible flood velocity for the site. For the HI-STORM FW system, Equation 4 governs since the coefficient of friction ( $\mu$ ) is less than the smallest value of  $D/h$  for the limiting overpack geometry (maximum height). The numerical value of  $V$  is computed as follows:

From Tables 3.2.1 and 3.2.2 and the drawings in Section 1.5, the diameter and maximum height of the overpack are

$$D = 139 \text{ in} = 11.6 \text{ ft}$$

$$h = 240 \text{ in} = 20.0 \text{ ft}$$

From Tables 3.2.3 and 3.2.5, the minimum weight of the HI-STORM FW overpack with an empty MPC (based on Ref. PWR fuel length and 150 pcf concrete density) is

$$W = 254,600 \text{ lbf}$$

Finally, assuming that  $W^* = 0.87W$ , the acceptable upper bound flood velocity is determined from Equation 4 as

$$V = \sqrt{\frac{1.876(0.53)(0.87 \times 254,600)}{(11.6)(20.0)}} = 30.8 \text{ ft/sec}$$

#### 3.4.4.1.2 Load Case 2: Design Basis Earthquake

In Subsection 2.2.3 (g), the combination of vertical and horizontal ZPA of the earthquake that would cause incipient loss of kinematic stability is derived using static equilibrium. The resulting inequality defines the threshold of the so-called low intensity earthquake for which the HI-STORM FW system is qualified without a dynamic analysis. However, an earthquake is a cyclic loading event which would produce rattling of the MPC inside the overpack and possibly large strains in the Confinement Boundary at the location of rattling impact between the MPC and the overpack guide tubes.

For earthquakes stronger than that defined by the inequalities in Subsection 2.2.3(g), it is necessary to perform a dynamic analysis. The dynamic stability analysis may be performed using either one of the following two approaches:

- i. Using the nomographs developed in NUREG/CR-6865 [3.4.7] to predict the cask rotation and sliding.
- ii. Performing a time history analysis for the cask modeled with 6 degrees-of-freedom and subjected to 3-dimensional seismic accelerations.

The first approach, although limited in its applications, is simple and conservative for the seismic stability evaluation of the HI-STORM FW storage cask as explained below. The nomograph developed in NUREG/CR-6865 [3.4.7] for cylindrical casks are based on extensive parametric study of the seismic response of HI-STORM 100 with a series of seismic inputs fitting three different spectral shapes. The seismic response is predicted through transient finite element analyses where the cask is supported on a flexible concrete pad founded on three substrates ranging from soft soil to rock. The NUREG study offers two sets of nomographs depending on the match of the site-specific free field horizontal spectrum with the three spectral shapes utilized in the study (after normalization to the Peak (Zero Period) Ground Acceleration (PGA)). The power law for the HI-STORM 100 response "y" (either peak cask top displacement or peak cask rotation) in terms of the ground motion parameter "x" at confidence band "m" standard deviations above the median response is:

$$y = Ax^B \exp(m S_{Y|x})$$

In the above equation, "A" and "B" are the nomograph curve fitting parameters, and "S" is the conditional standard deviation of the result data after undergoing a logarithmic transformation. The value for "m" is 0 (for the median curve), +1 (for the 84% confidence level) and -1 (for the 16%

confidence level). The units of “A” are meters (for displacement) and degrees (for rotation). The values for the coefficients are given below, as reproduced from [3.4.7]. The nomograph parameters are affected by the cask/pad coefficient of friction, but are independent of substrate stiffness.

Curve Fitting Parameters for Cylindrical Cask, NUREG/CR-0098 Earthquakes, PGA

	A (disp.)	B (disp.)	$S_{yx}$ (disp.)	A (rot.)	B (rot.)	$S_{yx}$ (rot.)
$\mu=0.2$	0.216	2.60	0.409	0.0217	0.689	0.718
$\mu=0.55$	0.911	4.06	0.814	6.70	3.94	0.794
$\mu=0.8$	1.150	4.16	0.796	9.01	4.09	0.765

Curve Fitting Parameters for Cylindrical Cask, Regulatory Guide 1.60 Earthquakes, PGA

	A (disp.)	B (disp.)	$S_{yx}$ (disp.)	A (rot.)	B (rot.)	$S_{yx}$ (rot.)
$\mu=0.2$	0.837	2.52	0.465	0.0733	1.71	0.785
$\mu=0.55$	8.96	4.80	1.03	62.5	4.71	0.956
$\mu=0.8$	15.4	5.04	1.13	114	4.94	1.12

Curve Fitting Parameters for Cylindrical Cask, NUREG/CR-6728 Earthquakes, PGA

	A (disp.)	B (disp.)	$S_{yx}$ (disp.)	A (rot.)	B (rot.)	$S_{yx}$ (rot.)
$\mu=0.2$	0.0897	1.88	0.377	0.0456	1.17	0.777
$\mu=0.55$	0.219	2.63	0.543	1.64	2.53	0.583
$\mu=0.8$	0.253	2.71	0.631	2.11	2.68	0.606

Curve Fitting Parameters for Cylindrical Cask, All Spectral Shapes, 1 Hz PSA

	A (disp.)	B (disp.)	$S_{yx}$ (disp.)	A (rot.)	B (rot.)	$S_{yx}$ (rot.)
$\mu=0.2$	0.271	2.15	0.532	0.0335	0.769	0.91
$\mu=0.55$	0.979	3.20	1.07	7.07	3.10	1.04
$\mu=0.8$	1.29	3.31	1.11	10.1	3.25	1.09

The use of the above nomographs for HI-STORM FW seismic stability analysis is conservative, as long as the h/r ratio (h = height to cask centroid, r = radius of the cask at interface with the pad) of HI-STORM FW is smaller than that of HI-STORM 100 cask, which is true in most cases. The nomographs should not be used when the substrate characteristics indicate that liquefaction will occur under a seismic event [3.4.7]. The basic analysis procedure is as follows:

- Demonstrate that the cask h/r ratio is less than that of HI-STORM 100 cask. If this condition is not satisfied, this approach cannot be used for the seismic stability analysis.
- Evaluate the site-specific substrate data to ensure that the site-specific substrate is within the range considered in the NUREG and that there is no potential for soil liquefaction under a seismic event.
- Compare the site-specific horizontal free-field response spectrum for 5% damping with those employed in the NUREG (after normalizing the site-specific data to 1g).

- a. If the site-specific spectrum is a good match with one of the spectrums employed, then use the nomograph appropriate to the matched spectrum and site-specific input at the ZPA to predict cask displacement and rotation.
- b. If the site-specific spectrum is not a good match with any of the spectra, then use the nomograph developed for all spectra with site-specific input at 1 Hz and 5% damping to predict cask displacement and rotation.

If the previously described NUREG/CR-6865 approach is not appropriate to use for a specific ISFSI site, the second approach should be used to perform the seismic stability evaluation for HI-STORM FW casks. The time history analysis approach, which is free of the limitations associated with NUREG/CR-6865, was used and approved by the USNRC to demonstrate the seismic stability of HI-STORM 100 casks at the Private Fuel Storage ISFSI. The input seismic acceleration time histories shall meet the relevant requirements specified in the SRP 3.7.1 [3.4.8] and shall be baseline corrected.

Finally, a small clearance between the MPC and the MPC guide tubes may lead to a high localized strain in the region of the shell where impacts from rattling of the canister under a seismic event occur. The extent of local strain from impact is minimized by locating the guide tube in the vertical direction such that its impact footprint is aligned with the surface of the closure lid which has been shimmed to close the crevice between the lid and the shell. Thus the impact between the guide tubes and the MPC lid will occur at a location where the maximum damage to the MPC shell will be local denting in the region where it is buttressed by the edge of a (thick) MPC lid. Therefore, a through-wall damage of the MPC shell is not credible. Furthermore, the force of impact will evidently be greater in the non-mechanistic tip-over case. Therefore, the seismic impact case is designated as non-governing for the guide tube/MPC impact scenario.

#### 3.4.4.1.3 Load Case 3: Tornado-Borne Projectiles

During a tornado event, the HI-STORM FW overpack and the HI-TRAC VW are assumed to be subjected to a constant wind force. They are 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.

##### a. Large Missile

##### Overturning Analysis

The large tornado missile acting at the top region of the cask (HI-STORM FW or HI-TRAC VW) to produce maximum overturning effect (Table 3.1.1) is analyzed to determine whether the cask will remain stable. Because the site-specific large missile is apt to be different from the one analyzed herein, the method of analysis presented here will provide the means for the site-specific safety evaluation pursuant to 10CFR72.212.

The overturning analysis of the cask under the tornado wind load and large missile impact is performed by solving the 1-DOF equation of motion for the cask angular rotation, which is same methodology used in the HI-STORM 100 FSAR (Docket No. 72-1014). Specifically, the solution of the post-impact dynamics problem is obtained by solving the following equation of motion:

$$I_r \alpha = \left( -W_c \frac{a}{2} \right) + F_{\max} \left( \frac{L}{2} \right)$$

where:

- $I_r$  = cask moment of inertia about the pivot point
- $\alpha$  = angular acceleration of the cask
- $W_c$  = lower bound weight of the cask
- $a$  = diameter of cask at its base (see Figure 3.4.7)
- $F_{\max}$  = force on the cask due to tornado wind/instantaneous pressure drop
- $L$  = height of the cask (see Figure 3.4.7)

The impacting missile enters into the above through the post-strike angular velocity of the cask, which is the relevant initial condition for the cask equation of motion. The solution gives the post-impact position of the cask centroid as a function of time, which indicates whether the cask remains stable.

The following assumptions are made in the analysis:

- i. The cask is assumed to be a rigid solid cylinder, with uniform mass distribution. This assumption implies that the cask sustains no plastic deformation (i.e. no absorption of energy through plastic deformation of the cask occurs).
- ii. The angle of incidence of the missile is assumed to be such that its overturning effect on the cask is maximized (see Figure 3.4.7).
- iii. The analysis considers the maximum height cask per Tables 3.2.1 and 3.2.2. The missile is assumed to strike at the highest point of the cask (see Figure 3.4.7), again maximizing the overturning effect.
- iv. The cask is assumed to pivot about a point at the bottom of the base plate opposite the location of missile impact and the application of wind force in order to conservatively maximize the propensity for overturning (see Figure 3.4.7).
- v. Inelastic impact is assumed, with the missile velocity reduced to zero after impact. This assumption conservatively lets the missile impart the maximum amount of angular momentum to the cask, and it is in agreement with missile impact tests conducted by EPRI [3.4.14].

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- vi. The analysis is performed using the minimum loaded HI-STORM FW weight per Table 3.2.8. A lighter cask will tend to rotate further after the missile strike. The weight of the missile is not included in the total post-impact weight.
- vii. Planar motion of the cask is assumed; any loads from out-of-plane wind forces are neglected.
- viii. The drag coefficient for a cylinder in turbulent cross flow is used.
- ix. The missile and wind loads are assumed to be perfectly aligned in direction.

The results for the post-impact response of the HI-STORM FW overpack and the HI-TRAC VW transfer cask are summarized in Table 3.4.5. The table shows that both casks remain in a vertical upright position (i.e., no overturning) in the aftermath of a large missile impact. The complete details of the tornado wind and large missile impact analyses for the HI-STORM FW overpack and the HI-TRAC VW transfer cask are provided in Appendix 3.A.

#### Sliding Analysis

A conservative calculation of the extent of sliding of the HI-STORM FW overpack and the HI-TRAC VW cask due to the impact of a large missile (Table 2.2.5) and tornado wind (Table 2.2.4) is obtained using a common formulation as explained below. A more realistic impact simulation using LS-DYNA, with less bounding assumptions, has been used in Subsection 3.4.4.1.4 to qualify the HI-STORM overpack for a non-mechanistic tip over event. While it is not necessary for demonstrating adequate safety margins for this problem, an LS-DYNA analysis could also be used to calculate the sliding potential of the HI-STORM FW and HI-TRAC VW for a large missile impact. In what follows, both HI-STORM FW and HI-TRAC VW are identified by the generic term "cask".

The principal assumptions that render these calculations for sliding conservative are:

- i. The weight of the cask used in the analysis is assumed to be the lowest per Table 3.2.8.
- ii. The cask is assumed to absorb the energy of impact purely by sliding. In other words, none of the impact energy is dissipated by the noise from the impact, from local plastic deformation in the cask at the location of impact, or from the potential tipping action of the cask.
- iii. The missile impact and high wind, which applies a steady drag force on the cask, are assumed to act synergistically to maximize the movement of the cask.
- iv. The cask is assumed to be freestanding on a concrete surface. The interface friction coefficient is assumed to be equal to that endorsed in the HI-STORM 100 FSAR (USNRC Docket No. 72-1014) and adopted here in the HI-STORM FW FSAR.

- v. The dynamic effect of the impact is represented by the force-time curve developed in the Bechtel topical report “Design of Structures for Missile Impact” [3.4.9], previously used to qualify the HI-STORM 100 System (USNCR Docket No. 72-1014).

The analysis for sliding under the above assumptions reduces to solving Newton's equation of motion of the form:

$$m \frac{d^2 x}{dt^2} = F(t) + F_{dp} - \mu mg$$

where

$m$ : mass of the cask,

$t$ : time coordinate with its origin set at the instant when the sum of the missile impact force and wind drag force overcomes the static friction force,

$x$ : displacement as a function of time coordinate  $t$ ,

$F(t)$ : missile impact force as a function of time (from [3.4.9]),

$F_{dp}$ : drag force from high wind,

$\mu$ : interface friction set as 0.53 for freestanding cask on a reinforced concrete pad in Docket No. 72-1014,

$g$ : acceleration due to gravity.

The above second-order differential equation is solved numerically in [3.4.15] for the HI-STORM FW overpack and the HI-TRAC VW transfer cask, and the calculated sliding displacements are summarized in Table 3.4.16.

Referring to the spacing dimensions for HI-STORM FW arrays in Table 1.4.1, the minimum space between HI-STORM FW overpacks and the minimum distance of the overpack to the edge of the pad are calculated. The above table demonstrates the HI-STORM FW overpack will not collide with another overpack, and the overpack will not slide off the pad due to the combined effects of a large tornado missile impact and high wind.

No generic limits for sliding are established for the HI-TRAC VW. Therefore, the sliding result for the HI-TRAC VW transfer cask in Table 3.4.16 is strictly informational.

#### b. Small and Intermediate Missiles

The small and intermediate missiles (Table 2.2.5) are analyzed to determine the extent to which they will penetrate the HI-STORM FW overpack or the HI-TRAC VW and cause potential damage to the MPC Enclosure Vessel. Classical energy balance methods are used to compute the depth of

penetration at the following impact locations:

- on the HI-STORM FW outer shell (with concrete backing)
- on the HI-STORM FW lid top plate (with concrete backing)
- on the HI-TRAC VW outer shell (with lead backing)
- on the top surface of the MPC upper lid

The MPC upper lid is analyzed for a direct missile impact because, when the MPC is placed inside the HI-TRAC VW, the MPC lid is theoretically accessible to a vertically downward directed small or intermediate missile.

The following assumptions are made in the analysis:

- i. The intermediate missile and the small missile are assumed to be unyielding, and hence the entire initial kinetic energy is assumed to be absorbed by local yielding and denting of the cask surface.
- ii. No credit is taken for the missile resistance offered by the HI-TRAC VW water jacket shell. It is assumed a priori that the small and intermediate missiles will penetrate the water jacket shell (with no energy loss). Therefore, in the analysis 100% of the missile impact energy is applied directly to the HI-TRAC VW outer shell.
- iii. For missile strikes on the side and top lid of the overpack, the analysis credits the structural resistance in compression offered by the concrete material that backs the outer shell and the lid.
- iv. The resistance from the concrete is conservatively assumed to act over an area equal to the target area of impact. In other words, no diffusion of the load is assumed to occur through the concrete.

The analyses documented in Appendix 3.B show that the depth of penetration of the small missile is less than the thinnest section of material on the exterior surface of the HI-STORM FW or the HI-TRAC VW. Therefore, the small missile will dent, but not penetrate, the cask. The 1-inch missile can enter the air inlet/outlet vents in the HI-STORM FW overpack, but geometry prevents a direct impact with the MPC.

For the intermediate missile, the analyses documented in Appendix 3.B show that there will be no penetration through the concrete surrounding the inner shell of the storage overpack or penetration of the top lid. Likewise, the intermediate missile will not penetrate the lead surrounding the HI-TRAC VW inner shell. Therefore, there will be no impairment to the Confinement Boundary due to tornado-borne missile strikes. Furthermore, since the HI-STORM FW and HI-TRAC VW inner shells are not compromised by the missile strike, there will be no permanent deformation of the inner shells and ready retrievability of the MPC will be assured.



The penetration results for the small and intermediate missile are summarized in Table 3.4.6.

#### 3.4.4.1.4 Load Case 4: Non-Mechanistic Tipover

The non-mechanistic tipover event, as described in Subsection 2.2.3(b), is site-dependent only to the extent that the stiffness of the target (ISFSI pad) affects the severity of the impact impulse. To bound the majority of ISFSI pad sites, the tipover analyses are performed using a stiff target foundation, which is defined in Table 2.2.9. The objectives of the analyses are 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 storage system. Furthermore, the maximum lateral deflection of the lateral surface of the fuel basket is within the limit assumed in the criticality analyses (Chapter 6), and therefore, the lateral deflection does not have an adverse effect on criticality safety.

The tipover event is an artificial construct wherein the HI-STORM FW overpack is assumed to be perched on its edge with its C.G. directly over the pivot point A (Figure 3.4.8). In this orientation, the overpack begins its downward rotation with zero initial velocity. Towards the end of the tipover, the overpack is horizontal with its downward velocity ranging from zero at the pivot point (point A) to a maximum at the farthest point of impact. The angular velocity at the instant of impact defines the downward velocity distribution along the contact line.

In the following, an explicit expression for calculating the angular velocity of the cask at the instant when it impacts on the ISFSI pad is derived. Referring to Figure 3.4.8, let  $r$  be the length AC where C is the cask centroid. Therefore,

$$r = \left( \frac{d^2}{4} + h^2 \right)^{1/2}$$

The mass moment of inertia of the HI-STORM FW system, considered as a rigid body, can be written about an axis through point A, as

$$I_A = I_c + \frac{W}{g} r^2$$

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 the ISFSI pad, is

$$I_A \frac{d^2 \theta_1}{dt^2} = Mgr \sin \theta_1$$

This equation can be rewritten in the form

$$\frac{I_A}{2} \frac{d(\dot{\theta}_1)^2}{d\theta_1} = Mgr \sin \theta_1$$

which can be integrated over the limits  $\theta_1 = 0$  to  $\theta_1 = \theta_{2f}$  (Figure 3.4.8). The final angular velocity  $\dot{\theta}_1$  at the time instant just prior to contact with the ISFSI pad is given by the expression

$$\dot{\theta}_1(t_B) = \sqrt{\frac{2 Mgr}{I_A} (1 - \cos \theta_{2f})}$$

where, from Figure 3.4.8,

$$\theta_{2f} = \cos^{-1} \left( \frac{d}{2r} \right)$$

This equation establishes the initial conditions for the final phase of the tip-over 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-STORM FW (Table 3.4.11) and the above equations, the angular velocity of impact is calculated as

$$\dot{\theta}_1(t_B) = 1.45 \text{ rad/sec}$$

The LS-DYNA analysis to characterize the response of the HI-STORM FW system under the non-mechanistic tipover event is focused on two principal demonstrations, namely:

- (i) The lateral deformation of the basket panels in the active fuel region is less than the limiting value in Table 2.2.11.
- (ii) The impact between the MPC guide tubes and the MPC does not cause a thru-wall penetration of the MPC shell.

Two LS-DYNA finite element models are developed to simulate the postulated tipover event of HI-STORM FW storage cask with loaded MPC-37 and MPC-89, respectively. The two LS-DYNA models are constructed according to the dimensions specified in the licensing drawings included in

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Section 1.5; the tallest configuration for each MPC type is considered to ensure a bounding tipover analysis. Because of geometric and loading symmetries, a half model of the loaded cask and impact target (i.e., the ISFSI pad) is considered in the analysis. The LS-DYNA models of the HI-STORM FW overpack and the MPC are described in Subsections 3.1.3.1 and 3.1.3.2, respectively.

The ISFSI pad LS-DYNA model, which consists of a 320"×100"×36" concrete pad and the underlying subgrade (800"×275"×470" in size) with non-reflective lateral and bottom surface boundaries, is identical to that used in the HI-STORM 100 tipover analysis documented in the HI-STORM 100 FSAR [3.1.4]. All structural members of the loaded cask are explicitly modeled so that any violation of the acceptance criteria can be found by examining the LS-DYNA simulation results (note: the fuel assembly, which is not expected to fail in a tipover event, is modeled as an elastic rectangular body). This is an improvement compared with the approach taken in the HI-STORM 100 tipover analysis, where the loaded MPC was modeled as a cylinder and therefore the structural integrity of the MPC and fuel basket had to be analyzed separately based on the rigid body deceleration result of the cask. Except for the fuel basket, which is divided into four parts based on the temperature distribution of the basket, each structural member of the cask is modeled as an independent part in the LS-DYNA model. Note that the critical weld connection between the MPC shell and the MPC lid is treated as a separate part and modeled with solid elements. Each of the two LS-DYNA models consists of forty-two parts, which are discretized with sufficiently high mesh density; very fine grids are used in modeling the MPC enclosure vessel, especially in the areas where high stress gradients are expected (e.g., initial impact location with the overpack). To ensure numerical accuracy, full integration thin shell and thick shell elements with 10 through-thickness integration points or multi-layer solid elements are used. The LS-DYNA tipover model consists of over 470,000 nodes and 255,000 elements for HI-STORM FW with loaded MPC-37, and the model for the cask with loaded MPC-89 consists of over 689,000 nodes and 350,000 elements.

The same ISFSI concrete pad material model used for the HI-STORM 100 tipover analysis reported in [3.1.4] is repeated for the HI-STORM FW tipover analysis. Specifically, the concrete pad behavior is characterized using the same LS-DYNA material model (i.e., MAT\_PSEUDO\_TENSOR or MAT\_016) as for the end drop and tipover analyses of the HI-STORM 100 storage cask (the only difference between the HI-STORM FW reference ISFSI concrete pad model and the model of the HI-STORM 100 Set B ISFSI concrete pad is thickness). Moreover, the subgrade is also conservatively modeled as an elastic material as before. Note that this ISFSI pad material modeling approach was originally taken in the USNRC approved storage cask tipover and end drop LS-DYNA analyses [3.4.5] where a good correlation was obtained between the analysis results and the test results.

To assess the potential damage of the cask caused by the tipover accident, an LS-DYNA nonlinear material model with strain rate effect is used to model the responses of all HI-STORM FW cask structural members based on the true stress-strain curves of the corresponding materials. Note that the strain rate effect for the fuel basket material, i.e., Metamic HT, is not considered for conservatism.

Figures 3.4.9 to 3.4.14 depict the two finite-element tipover analysis models developed for the bounding HI-STORM FW cask configurations with loaded MPC-37 and MPC-89, respectively.

As shown in Figure 3.4.15, the fuel basket does not experience significant plastic deformation in the active fuel region to exceed the acceptable limits; plastic deformation is essentially limited locally in cells near the top of the basket beyond the active fuel region for both MPC-37 and MPC-89 baskets. Note that the basket corner welds are not considered in the tip-over analysis for conservatism.. The fuel basket is considered to be structurally safe since it can continue maintaining appropriate spacing between fuel assemblies after the tipover event. The MPC enclosure vessel experiences minor plastic deformation at the impact locations with the overpack guide tubes; the maximum local plastic strain (9.9%, see Figure 3.4.16) is well below the failure strain of the material and smaller than the plastic strain limit (i.e., at least 0.2 for stainless steel) recommended by [3.4.6] for ASME NB components. Similarly, local plastic deformation occurs in the overpack shear ring near the cask-to-pad impact location as shown in Figure 3.4.17. However, the shielding capacity of overpack will not be compromised by the tipover accident and there is no gross plastic deformation in the overpack inner shell to affect the retrievability of the MPC. In addition, the cask closure lid bolts are demonstrated to be structurally safe after the tipover event, only a negligibly small plastic strain is observed in the bolt near the impact location (see Figure 3.4.18). Therefore, the cask lid will not dislodge after the tipover event. Finally, Figures 3.4.19 and 3.4.20 present the deceleration time history results of the cask lid predicted by LS-DYNA. The peak rigid body decelerations, measured for the HI-STORM FW lid concrete, are shown to be 61.75 g's in the vertical direction and 16.71 g's in the horizontal direction, respectively. Note that the deceleration time histories are filtered using the LS-DYNA built-in Butterworth filter with a cut-off frequency of 350 Hz; the same filter was used for the HI-STORM 100 non-mechanistic tipover analysis [3.1.4].

The structural integrity of the HI-STORM FW lid cannot be ascertained from the LS-DYNA tipover analyses since some components of the lid, namely the lid outer shell and the lid gussets, are defined as rigid members in order to simplify the modeling effort and maintain proper connectivity. Therefore, a separate tipover analysis has been performed for the HI-STORM FW lid using ANSYS, wherein a bounding peak rigid body deceleration established based on LS-DYNA tipover analysis results is statically applied to the lid. The finite element model is identical to the one used in Subsection 3.4.3 to simulate a vertical lift of the HI-STORM FW lid (Figure 3.4.5), except that the eight circumferential gussets are conservatively neglected (i.e., deleted from the finite element model).

The resulting stress distribution in the HI-STORM FW lid is shown in Figure 3.4.21. Per Subsection 2.2.3, the HI-STORM FW lid should not suffer any gross loss of shielding as a result of the non-mechanistic tipover event. To satisfy this criterion, the primary membrane stresses in the lid components are compared against the material yield strength. The most heavily loaded component is the upper shim plate closest to the point of impact (Figure 3.4.21). In order to determine the primary membrane stress in the upper shim plate, the stresses are linearized along a path that follows the outside vertical edge of the upper shim plate (see Figure 3.4.21 for path definition). Figure 3.4.22 shows the linearized stress results. Since the membrane stress is less than the yield strength of the material at 300°F (Table 3.3.6), it is concluded that the lid will not suffer any gross loss of

shielding as a result of the non-mechanistic tipover event. The complete details of the lid tipover analysis are provided in [3.4.13].

Finally, to evaluate the potential for crack propagation and growth for the MPC fuel baskets under the non-mechanistic tipover event, a conservative crack propagation analysis is carried out for both MPC-37 and MPC-89 fuel baskets using the same methodology utilized in Attachment D of [1.2.6] to evaluate the HI-STAR 180 F-37 fuel basket in support of the HI-STAR 180 SAR [3.1.10]. The crack propagation analysis is bounding since the maximum tensile strength of the basket material (28.2 ksi) documented in Table 1.2.8 is conservatively considered as the maximum tensile stress experienced by the Metamic fuel baskets in the tip-over accident and used as input to the following crack propagation analysis.

Per [1.2.6] the critical stress intensity factor of Metamic-HT panels is estimated to be

$$K_{IC} = 30\text{ksi}\sqrt{\text{in}}$$

based on Charpy V-notch absorbed energy (CVE) correlations for steels. The estimated value is consistent with the range for aluminum alloys, which is 20 to 50  $\text{MPa}\sqrt{\text{m}}$  or 18.2 to 45  $\text{ksi}\sqrt{\text{in}}$  per Table 3 of [3.4.19]. Next the minimum crack size,  $a_{\min}$ , for crack propagation to occur is calculated below using the formula for a through-thickness edge crack given in [3.1.5]. Although the formula is derived for a straight-edge specimen, the use of the maximum tensile strength of the fuel basket material as the maximum tensile stress experienced by the basket well compensates for the geometric difference between the basket panel and the specimen. Moreover, the maximum size of a pre-existing crack (1/16") in the fuel basket panel is less than 1/9th of the basket panel thickness (0.59"). Thus, the assumption of a through-thickness edge crack is very conservative. The result is

$$a_{\min} = \frac{\left(\frac{K_{IC}}{1.12\sigma_{\max}}\right)^2}{\pi} = \frac{\left[\frac{30\text{ksi}\sqrt{\text{in}}}{1.12(28.2\text{ksi})}\right]^2}{\pi} = 0.287\text{in}$$

And the safety factor against crack propagation (based on a 1/16" minimum detectable flaw size) is

$$SF = \frac{a_{\min}}{a_{\det}} = \frac{0.287\text{in}}{0.0625\text{in}} = 4.595$$

The calculated minimum crack size is about 4.6 times the maximum possible pre-existing crack size in the fuel basket (based on 100% surface inspection of each panel). The large safety factor ensures that crack propagation in the HI-STORM FW fuel baskets will not occur due to the non-mechanistic tipover event.

#### 3.4.4.1.5 Load Case 5: Design Internal Pressure

The MPC Enclosure Vessel, which is designed to meet the stress intensity limits of ASME Subsection NB [3.4.4], is analyzed for design internal pressure (Table 2.2.1) using the ANSYS finite element code [3.4.1]. Except for the applied loads and the boundary conditions, the finite element model of the MPC Enclosure Vessel used for this load case is identical to the model described in Subsections 3.1.3.2 and 3.4.3.2 for the MPC lifting analysis.

The only load applied to the finite element model for this load case is the MPC design internal pressure for normal conditions (Table 2.2.1). All internal surfaces of the MPC storage cavity are subjected to the design pressure. The center node on the top surface of the MPC upper lid is fixed against translation in all directions. Symmetric boundary conditions are applied to the two vertical symmetry planes. This set of boundary conditions allows the MPC Enclosure Vessel to deform freely under the applied pressure load. Figure 3.4.31 graphically depicts the applied pressure load and the boundary conditions for Load Case 5.

The stress intensity distribution in the MPC Enclosure Vessel under design internal pressure is shown in Figure 3.4.23. Figures 3.4.32 and 3.4.33 plot the thru-thickness variation of the stress intensity at the baseplate center and at the baseplate-to-shell juncture, respectively. The maximum primary and secondary stress intensities in the MPC Enclosure Vessel are compared with the applicable stress intensity limits from Subsection NB of the ASME Code. The allowable stress intensities are taken at 450°F for the MPC shell and MPC lids, 300°F for the baseplate, and 250°F at the baseplate-to-shell juncture. The maximum calculated stress intensities in the MPC Enclosure Vessel, and their corresponding allowable limits, are summarized in Table 3.4.7 for Load Case 5.

Since the stress intensity distribution in the MPC Enclosure Vessel is a linear function of the internal pressure, and the stress intensity limits for normal and off-normal conditions are the same (Table 3.1.7), the minimum calculated safety factor from Table 3.4.7 is used to establish the internal pressure limit for off-normal conditions (Table 2.2.1).

#### 3.4.4.1.6 Load Case 6: Maximum Internal Pressure Under Accident Conditions

The maximum pressure in the MPC Enclosure Vessel under accident conditions is specified in Table 2.2.1. The stress analysis under this pressure condition uses the same model as the one described in the preceding subsection. The only change is the magnitude of the applied pressure. Figure 3.4.34 graphically depicts the applied pressure load and the boundary conditions for Load Case 6.

The stress intensity distribution in the MPC Enclosure Vessel under accident internal pressure is shown in Figure 3.4.24. The maximum primary stress intensities in the MPC Enclosure Vessel are compared with the applicable stress intensity limits from Subsection NB of the ASME Code [3.4.4]. The allowable stress intensities are taken at 450°F for the MPC shell and MPC lids, 300°F for the baseplate, and 250°F at the baseplate-to-shell juncture. These temperatures bound the calculated temperatures under normal operating conditions for the respective MPC components based on the thermal evaluations in Chapter 4. The allowable stress intensities are determined based on normal operating temperatures since the MPC accident internal pressure is dictated by the 100% fuel rod rupture accident, which does not cause any significant rise in MPC temperatures. In fact, the

temperatures inside the MPC tend to decrease as a result of the 100% fuel rod rupture accident due to the increase in the density and internal pressure of the circulating gas. The maximum calculated stress intensities in the MPC Enclosure Vessel, and their corresponding allowable limits, are summarized in Table 3.4.8 for Load Case 6.

#### 3.4.4.1.7 Load Case 7: Accident External Pressure

The only affected component for this load case is the MPC Enclosure Vessel. The accident external pressure (Table 2.2.1) is selected sufficiently high to envelop hydraulic-pressure in the case of flood or explosion-induced pressure at all ISFSI Sites.

The main effect of an external pressure on the MPC is to cause compressive stress in the MPC shell. Therefore, the potential of buckling must be investigated. The methodology used for this investigation is from ASME Code Case N-284-2 (Metal Containment Shell Buckling Design Methods, Section III, Division 1, Class MC (1/07)). This Code Case has been previously used by Holtec in [3.1.4] and accepted by the NRC as a valid method for evaluation of stability in vessels.

The detailed evaluation of the MPC shell under accident external pressure is provided in Appendix 3.C. It is concluded that positive safety margins exist so that elastic or plastic instability of the maximum height MPC shell does not occur under the applied pressure.

#### 3.4.4.1.8 Load Case 8: Non-Mechanistic Heat-Up of the HI-TRAC VW Water Jacket

Even though the analyses presented in Chapter 4 indicate that the temperature of water in the water jacket shall not reach boiling and the rupture disks will not open, it is (non-mechanistically) assumed that the hydraulic pressure in the water jacket reaches the relief devices' set point. The object of this analysis is to demonstrate that the stresses in the water jacket and its welds shall be below the limits set down in an appropriate reference ASME Boiler and Pressure Vessel Code (Section II Class 3) for the Level D service condition. The accident pressure inside the water jacket is given in Table 2.2.1.

The HI-TRAC VW water jacket is analyzed using classical strength-of-materials. Specifically, the unsupported span of the water jacket shell between radial ribs is treated as a curved beam, with clamped ends, under a uniformly distributed radial pressure. The force and moment reactions at the ends of the curved beam for this type of loading are calculated using the formula for Case 5j of Table 18 in [3.4.16]. The primary membrane plus bending stress is then calculated using the formula for Case 1 of Table 16 in [3.4.16]. Figure 3.4.35 depicts the curved beam model that is used to analyze the water jacket shell and defines the key input variables. The input values that are used in the calculations are provided in Table 3.4.12.

The bottom flange, which serves as the base of the water jacket, is conservatively analyzed as an annular plate clamped at the water jacket inside diameter and simply supported at the water jacket outside diameter. The maximum bending stress in the bottom flange is calculated using the following formula from [3.4.18, Art. 23]:

$$\sigma_{\max} = k \frac{q \cdot a^2}{h^2}$$

where  $q$  is the internal pressure inside the water jacket (= 73.65 psi),  $a$  is the outside radius of the water jacket (= 47.5 in), and  $h$  is the thickness of the bottom flange (= 2.0 in). The analyzed pressure accounts for the accident internal pressure inside the water jacket (Table 2.2.1) plus the hydrostatic pressure at the base of the water jacket. The value of  $k$  is dependent on the diameter ratio of the annular plate and the boundary conditions. Per Table 5 of [3.4.18],  $k$  is equal to 0.122 for a bounding diameter ratio of 1.25 and simply supported-clamped boundary conditions (Case 4). Therefore, the maximum bending stress in the bottom flange is:

$$\sigma_{\max} = 5,068 \text{ psi}$$

Per Table 3.1.6, the allowable primary membrane plus bending stress intensity for SA-516 Gr. 70 material (at 400°F) is 58,500 psi, which means the factor of safety is greater than 10.

The maximum stresses in the various water jacket components, including the connecting welds, are summarized in Table 3.4.9.

#### 3.4.4.1.9 Load Case 9: Handling of Components

The stress analyses of the MPC, the HI-STORM FW overpack, and the HI-TRAC VW transfer cask under normal handling conditions are presented in Subsection 3.4.3.

#### 3.4.4.1.10 Load Case 10: Snow Load

In accordance with Table 3.1.1, the HI-STORM FW lid is analyzed using ANSYS to demonstrate that the design basis snow load (Table 2.2.8) does not cause stress levels in the overpack lid to exceed ASME Subsection NF stress limits for Level A. The finite element model is identical to the one used in Subsection 3.4.3 to simulate a vertical lift of the HI-STORM FW lid (see Figure 3.4.5). For conservatism, a pressure load of 10 psig is used in the finite element analysis. The stress distribution in the lid under the bounding snow load is shown in Figure 3.4.25. The maximum stress results are summarized in Table 3.4.10. For conservatism, the maximum calculated stress at any point on the lid, including secondary stress contributions, is compared against the primary membrane and primary bending stress limits per ASME Subsection NF.



#### 3.4.4.1.11 Load Case 11: MPC Reflood Event

During a MPC reflood event, water is introduced to the MPC cavity through the lid drain line to cooldown the MPC internals and support fuel unloading. This quenching operation induces thermal stresses and strains in the fuel rod cladding, which are maximum at the boundary interface between the rising water and the dry (gaseous) cavity. The following analysis demonstrates that the maximum total strain in the fuel cladding due to the reflood event is well below the failure strain limit of the material. Thus, the fuel rod cladding will not be breached due to the MPC reflood event.

The analysis is carried out using the finite element code ANSYS [3.4.1]. The model, which is shown in Figure 3.4.37, is constructed using 4-node plastic large strain elements (SHELL43) based on the cladding dimensions of the PWR reference fuel type. The overall length of the model is equal to 30 times the outside diameter of the fuel cladding. As seen in Figure 3.4.37, the mesh size is reduced at the boundary between the wetted fuel rod and the dry fuel rod, where the highest stresses and strains occur. To account for the gas pressure inside the fuel rod, the top end of the fuel rod is fixed in the vertical direction, and an equivalent axial force is applied at the bottom end. A radial pressure is also applied to the inside surface of the fuel cladding (see Figure 3.4.38). The fuel cladding material is modeled as a bi-linear isotropic hardening material with temperature dependent properties. The key input data used to develop the finite element model are summarized in Table 3.4.14.

The MPC reflood pressure, which is restricted to below the normal condition pressure limit, is too low to have an adverse effect on the fuel cladding, the reflood water pressure acts to produce compressive hoop stresses which help reduce the tensile hoop stress (albeit by a small amount) from the internal gas pressure in the rods. Therefore, the MPC flooding pressure has no harmful consequence to the fuel cladding and is neglected in the analysis.

At  $t = 0$  sec, the uniform temperature throughout the entire fuel rod is set at 752°F (400°C), which equals the fuel cladding temperature limit under normal operating conditions. At  $t = 0.1$  sec, the temperature assigned to the lower half of the fuel rod model is suddenly reduced to 80°F to simulate the water quenching (see Figure 3.4.39). The resulting stress and strain distributions in the fuel rod are shown in Figures 3.4.40 and 3.4.41, respectively. The maximum stress and strain values are summarized in Table 3.4.15. The maximum total strain in the fuel rod is well below the failure strain limit of 1.7% for the cladding material per [3.4.20]. In fact, the maximum stress and strain in the fuel rod remain in the elastic range.

The analysis described above makes a number of assumptions that significantly overstate the computed thru-wall strain in the fuel cladding. The major assumptions are:

1. Even though the peak cladding temperature occurs at a localized location, the fuel rod is modeled as a pressurized tube with closed ends at a uniform temperature that is greater than the maximum peak cladding temperature value reported in Chapter 4 when the MPC is in the HI-TRAC under the Design Basis heat load condition.
2. The rapid thermal straining of the pressurized tube (fuel rod) due to the quenching effect of water is simulated as a step transient wherein the temperature of the quenched portion of the

tube is assumed to drop down to the injected water temperature (assumed to be 80°F) causing a step change in the cladding wall temperature in the longitudinal direction at its interface with the “dry” portion of the tube. This assumption is extremely conservative because in actuality the immersed portion of the fuel rod is blanketed by vapor which acts to retard the severity of the thermal transient.

3. Even though, as the rod is gradually immersed in water, the axial heat conduction will tend to cool the un-immersed portion of the tube thus reducing the  $\Delta T$  at the quenched/dry interface, no credit for axial conduction is taken.
4. The cooling of the fuel rod by gradual immersion in the water has the beneficial effect of reducing the internal pressure (per the ideal gas law) and thus the magnitude of pressure induced stress in the fuel cladding. As the peak cladding temperature in the MPC is reached in the upper half of the fuel rods (see Chapter 4), a substantial amount of rod is cooled by water (as its level gradually rises inside the MPC) before the vulnerable zone (where the peak cladding temperature exists) is subjected to the thermal transient from quenching. No credit for this amelioration of the pressure stresses due to the gradual cooling of the rod is taken in the analysis.

In summary, even though the analysis presented above is highly conservative, the maximum stress and strain in the fuel rod remain elastic. Moreover, the maximum strain is less than the failure strain limit by a factor of 6. Thus, the MPC reflood event will not cause a breach of the fuel rod cladding.

### 3.4.5 Cold

A discussion of the resistance to failure due to brittle fracture is provided in Subsection 3.1.2.

The value of the ambient temperature has two principal effects on the HI-STORM FW 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 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 –40 degree F ambient would be smaller than the “heat” condition of storage, treated in the preceding subsection. Additionally, the allowable stress limits tend to increase as the component temperatures decrease.

Therefore, the stress margins computed in Subsection 3.4.4 can be conservatively assumed to apply to the "cold" condition as well.

Finally, it can be readily shown that the HI-STORM FW system is engineered to withstand "cold" temperatures (-40 degrees F) without impairment of its storage function.

Unlike the MPC, the HI-STORM FW storage overpack is an open structure; it contains no pressure. Its stress field is unaffected by the ambient temperature, unless low temperatures produce brittle fracture due to the small stresses which develop from self-weight of the structure and from the minute difference in the thermal expansion coefficients in the constituent parts of the equipment (steel and concrete). To prevent brittle fracture, all steel material in HI-STORM FW is qualified by impact testing pursuant to the ASME Code (Table 3.1.9).

The structural material used in the MPC (Alloy X) is recognized to be completely immune from brittle fracture in the ASME Codes.

As no liquids are included in the HI-STORM FW storage overpack design, loads due to expansion of freezing liquids are not considered. The HI-TRAC VW transfer cask utilizes demineralized water in the water jacket. However, the specified lowest service temperature for the HI-TRAC VW is 0 degrees F and a 25% ethylene glycol solution is required for the temperatures from 0 degrees F to 32 degrees F. Therefore, loads due to expansion of freezing liquids are not considered.

There is one condition, however, that does require examination to ensure ready retrievability of the fuel. Under a postulated loading of an MPC from a HI-TRAC VW transfer cask into a cold HI-STORM FW storage overpack, it must be demonstrated that sufficient clearances are available to preclude interference when the "hot" MPC is inserted into a "cold" storage overpack. To this end, a bounding analysis for free thermal expansions has been performed in Subsection 4.4.6, wherein the MPC shell is postulated at its maximum design basis temperature and the thermal expansion of the overpack is ignored. The results from the evaluation of free thermal expansion are summarized in Table 4.4.6. The final radial clearance is sufficient to preclude jamming of the MPC upon insertion into a cold HI-STORM FW storage overpack.

### 3.4.6 Miscellaneous Evaluations

#### 3.4.6.1 Structural Integrity of Damaged Fuel Containers (DFCs)

The Damaged Fuel Container (DFC) is used to store fuel that is physically impaired such that it cannot be handled by normal means. The DFC, as shown in the licensing drawings, is equipped with a handle welded to a square cellular box with a perforated baseplate structurally capable of supporting the weight of the fuel while permitting water (but not particulates) to pass through. All load bearing members of the DFC are designed to meet Level A service limit when holding a spent fuel assembly.

Because the DFC is always handled under water, there are no radiation release-related issues

associated with it.

### 3.4.7 Service Life of HI-STORM FW and HI-TRAC VW

The term of the 10CFR72, Subpart L C of C, granted by the NRC is 20 years; therefore, the License Life (see Glossary) of all components is 20 years. Nonetheless, the HI-STORM FW storage overpack and the HI-TRAC VW transfer cask are engineered for 60 years of design life, while satisfying the conservative design requirements defined in Chapter 2, including the regulatory requirements of 10CFR72. In addition, the storage overpack and HI-TRAC VW are designed, fabricated, and inspected under the comprehensive Quality Assurance Program approved by the USNRC and in accordance with the applicable requirements of the ACI and ASME Codes. This assures high design margins, high quality fabrication, and verification of compliance through rigorous inspection and testing, as described in Chapter 10 and the licensing drawings in Section 1.5. Technical Specifications defined in Chapter 13 assure that the integrity of the cask and the contained MPC are maintained throughout the components' design life. The design life of a component, as defined in the Glossary, is the minimum duration for which the equipment or system is engineered to perform its intended function if operated and maintained in accordance with the FSAR. The design life is essentially the lower bound value of the service life, which is the expected functioning life of the component or system. Therefore, component longevity should be: licensed life < design life < service life. (The licensed life, enunciated by the USNRC, is the most pessimistic estimate of a component's life span). For purposes of further discussion, we principally focus on the service life of the HI-STORM FW system components that, as stated earlier, is the reasonable expectation of equipment's functioning life span.

The service life of the storage overpack and HI-TRAC VW transfer cask is further discussed in the following.

#### 3.4.7.1 Storage Overpack

The principal design considerations that bear on the adequacy of the storage overpack for the service life are addressed as follows:

##### Exposure to Environmental Effects

All exposed surfaces of the HI-STORM FW overpack are made from ferritic steels that are readily painted. Concrete, which serves strictly as a shielding material, is completely encased in steel. Therefore, the potential of environmental vagaries such as spalling of concrete, are ruled out for HI-STORM FW. Under normal storage conditions, the bulk temperature of the HI-STORM FW storage 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-STORM FW storage overpack. Similarly, corrosion of structural steel embedded in the concrete structures due to salinity in the environment at coastal sites is not a concern for HI-STORM FW because HI-STORM FW does not rely on rebars (indeed, it contains no rebars). As discussed in Appendix 1.D of HI-STORM 100 FSAR, the aggregates, cement and water used in the storage cask concrete are carefully

controlled to provide high durability and resistance to temperature effects. The configuration of the storage overpack assures resistance to freeze-thaw degradation. In addition, the storage overpack is specifically designed for a full range of enveloping design basis natural phenomena that could occur over the 60-year design life of the storage overpack as defined in Subsection 2.2.3 and evaluated in Chapter 12. Chapter 8 provides further discussions on chemical and galvanic reactions, material compatibility and operating environments.

### Material Degradation

As discussed in Chapter 8, the relatively low neutron flux to which the storage 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 storage overpack throughout the 60-year design life are defined in Chapter 10. These requirements include provisions for routine inspection of the storage overpack exterior and periodic visual verification that the ventilation flow paths of the storage overpack are free and clear of debris. ISFSIs located in areas subject to atmospheric conditions that 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-STORM FW system is designed for easy retrieval of the MPC from the storage overpack should it become necessary to perform more detailed inspections and repairs on the storage overpack.

The above findings are consistent with those of the NRC's Waste Confidence Decision Review [3.4.10], which concluded that dry storage systems designed, fabricated, inspected, and operated in accordance with such requirements are adequate for a 100-year service life while satisfying the requirements of 10CFR72.

### **3.4.7.2 Transfer Cask**

The principal design considerations that bear on the adequacy of the HI-TRAC VW transfer cask for the service life are addressed as follows:

#### Exposure to Environmental Effects

All transfer cask materials that come in contact with the spent fuel pool are coated to facilitate decontamination. The HI-TRAC VW is designed for repeated normal condition handling operations with high factor of safety to assure structural integrity. The resulting cyclic loading produces stresses that are well below the endurance limit of the cask's materials, and therefore, will not lead to a fatigue failure in the transfer cask. All other off-normal or postulated accident conditions are

infrequent or one-time occurrences that do not contribute significantly to fatigue. In addition, the transfer cask utilizes materials that are not susceptible to brittle fracture during the lowest temperature permitted for loading, as discussed in Subsection 8.4.3.

Chapter 8 provides further discussions on chemical and galvanic reactions, material compatibility and operating environments.

### Material Degradation

As discussed in Chapter 8, all transfer cask materials that are susceptible to corrosion are coated. The controlled environment in which the HI-TRAC VW is used mitigates damage due to direct exposure to corrosive chemicals that may be present in other industrial applications. The infrequent use and relatively low neutron flux to which the HI-TRAC VW materials are subjected do not result in radiation embrittlement or degradation of the HI-TRAC's shielding materials that could impair the HI-TRAC's intended safety function. The HI-TRAC VW transfer cask materials are selected for durability and wear resistance for their deployment.

### Maintenance and Inspection Provisions

The requirements for periodic inspection and maintenance of the HI-TRAC VW transfer cask throughout the 60-year design life are defined in Chapter 10. These requirements include provisions for routine inspection of the HI-TRAC VW transfer cask for damage prior to each use, including an annual inspection of the lifting attachments. Precautions are taken during lid handling operations to protect the sealing surfaces of the bottom lid. The leak tightness of the liquid neutron shield is verified periodically. The water jacket pressure rupture discs and other fittings used can be easily removed.

## 3.4.8 MPC Service Life

The term of the 10CFR72, Subpart L C of C, granted by the NRC (i.e., licensed life) is 20 years. Nonetheless, the HI-STORM FW MPCs are designed for 60 years of design life, 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 60-year life of the MPC is provided through the following:

- Design, fabrication, and inspection invoke the pertinent requirements of the ASME Code, as applicable, assures high inherent design margins in operating modes.
- Fabrication and inspection performed in accordance with the comprehensive Quality Assurance program assures competent compliance with the fabrication requirements.
- Use of materials with known characteristics, verified through rigorous inspection and testing, as described in Chapter 10, assures component compliance with design requirements.

- Use of welding procedures in full compliance with Section III of the ASME Code ensures high-quality weld joints.

Technical Specifications, as defined in Chapter 13, have been developed and imposed on the MPC that assure that the integrity of the MPC and the contained SNF assemblies are maintained throughout the 60-year design life of the MPC.

The principal design considerations bearing on the adequacy of the MPC for the 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 moisture in the MPC is removed to eliminate all oxidizing liquids and gases and the MPC cavity is 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 that is made from stainless alloy stock (endurance limit well in excess of 20,000 psi). 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 and the associated fabrication and closure requirements for the canister are specifically set down to assure that an inert helium atmosphere is maintained in the canister throughout the 60-year design 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 60-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 60-year design life of the MPC. Information on the characteristics of the borated neutron absorbing material used in the MPC fuel basket is provided in Subsection 1.2.1 and Chapter 8. The relatively low neutron flux, to which this borated material is subjected and will continue to decay over time, does not result in significant depletion of the material's available boron to perform 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 demonstrates that the boron depletion in the neutron absorber material is negligible over a 60-year duration. Thus, sufficient levels of boron are present in the fuel basket neutron absorbing material to maintain criticality safety functions over the 60-year design 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.

### 3.4.9 Design and Service Life

The discussion in the preceding sections seeks to provide the logical underpinnings for setting the design life of the storage overpacks, the HI-TRAC VW transfer cask, and the MPCs as sixty years. Design life, as stated earlier, is a lower bound value for the expected performance life of a component (service life). If operated and maintained in accordance with this Safety Analysis Report, Holtec International expects the service life of HI-STORM FW casks to substantially exceed their design life values.



Table 3.4.1			
STRESS INTENSITY RESULTS FOR MPC ENCLOSURE VESSEL – NORMAL HANDLING			
Item	Calculated Value (ksi)	Allowable Limit (ksi)	Safety Factor
Lid – Primary Membrane Stress Intensity	6.94	18.05	2.60
Lid – Local Membrane Plus Primary Bending Stress Intensity	6.94	27.1	3.90
Baseplate – Primary Membrane Stress Intensity	8.32	20.0	2.40
Baseplate – Local Membrane Plus Primary Bending Stress Intensity	21.8	30.0	1.38
Shell – Primary Membrane Stress Intensity	13.14	18.05	1.37
Shell – Local Membrane Plus Primary Bending Plus Secondary Stress Intensity	56.50	60.0	1.06

Table 3.4.2			
STRESS RESULTS FOR HI-TRAC VW – NORMAL HANDLING			
Item	Calculated Value (ksi)	Allowable Limit (ksi)	Safety Factor
Top Flange-to- Inner/Outer Shell Weld – Primary Shear Stress	5.79	17.4	3.01
Inner/Outer Shell – Primary Membrane Stress	1.72	19.6	11.4
Bottom Lid Bolts – Tensile Stress	9.28	57.5	6.20
Bottom Lid – Primary Bending Stress	3.80	30.0	7.90

Table 3.4.3			
STRESS RESULTS FOR HI-STORM FW – NORMAL HANDLING			
Item	Calculated Value (ksi)	Allowable Limit (ksi)	Safety Factor
Inner/Outer Shell – Primary Membrane Stress	1.92	20.0	10.4
Inner/Outer Shell – Primary Membrane Plus Bending Stress	3.42	30.0	8.77
Baseplate – Primary Membrane Stress	1.97	20.0	10.2
Baseplate – Primary Membrane Plus Bending Stress	3.42	30.0	8.77
Lifting Rib – Primary Membrane Stress	4.79	20.0	4.18
Lifting Rib – Primary Membrane Plus Bending Stress	6.22	30.0	4.82
Shell-to-Baseplate Weld – Primary Shear Stress	4.56	21.0	4.60

Table 3.4.4			
STRESS RESULTS FOR HI-STORM FW LID – NORMAL HANDLING			
Item	Calculated Value (ksi)	Allowable Limit (ksi)	Safety Factor
Maximum Primary Membrane Stress	1.57	16.6	10.6
Maximum Primary Membrane Plus Bending Stress	1.57	24.9	15.9

Table 3.4.5			
CASK ROTATIONS DUE TO LARGE MISSILE IMPACT			
Event	Calculated Value (deg)	Allowable Limit (deg)	Safety Factor
Missile Impact plus Tornado Wind on HI- STORM FW	3.83	30.3	7.91
Missile Impact plus Pressure Drop on HI- STORM FW	4.37	30.3	6.93
Missile Impact plus Tornado Wind on HI- TRAC VW	14.88	23.6	1.59
Missile Impact plus Pressure Drop on HI- TRAC VW	12.66	23.6	1.86

Table 3.4.6			
MISSILE PENETRATION RESULTS – SMALL AND INTERMEDIATE MISSILE			
Missile Type – Impact Location	Calculated Value (in)	Allowable Limit (in)	Safety Factor
Small Missile – All Impact Locations	< 0.4 in	> 0.5 in (MPC shell thickness) <sup>†</sup>	> 1.25
Intermediate Missile – Side Strike on HI-STORM FW Outer Shell (away from Inlet)	8.39	29.00	3.46
Intermediate Missile – Side Strike on HI-STORM FW Outer Shell (at Inlet)	11.69	24.00	2.05
Intermediate Missile – End Strike on HI-STORM FW Lid	10.46	19.25	1.84
Intermediate Missile – Side Strike on HI-TRAC VW Outer Shell	0.50	1.50	3.00
Intermediate Missile – End Strike on MPC Closure Lid	0.23	9.00	39.13

<sup>†</sup> In reality, a maximum velocity impact between the small projectile missile and the MPC shell is not credible due to the geometry of the HI-STORM FW inlet and outlet vents (i.e., no direct line of sight).

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Table 3.4.7			
STRESS INTENSITY RESULTS FOR MPC ENCLOSURE VESSEL – DESIGN INTERNAL PRESSURE			
Item	Calculated Value (ksi)	Allowable Limit (ksi)	Safety Factor
Lid – Primary Membrane Stress Intensity	5.98	18.05	3.02
Lid – Local Membrane Plus Primary Bending Stress Intensity	5.98	27.1	4.53
Baseplate – Primary Membrane Stress Intensity	7.12	20.0	2.81
Baseplate – Local Membrane Plus Primary Bending Stress Intensity	18.65	30.0	1.61
Shell – Primary Membrane Stress Intensity	11.50	18.05	1.57
Shell – Local Membrane Plus Primary Bending Plus Secondary Stress Intensity	50.10	60.0	1.20

Table 3.4.8			
STRESS INTENSITY RESULTS FOR MPC ENCLOSURE VESSEL – ACCIDENT INTERNAL PRESSURE			
Item	Calculated Value (ksi)	Allowable Limit (ksi)	Safety Factor
Lid – Primary Membrane Stress Intensity	11.97	43.3	3.62
Lid – Local Membrane Plus Primary Bending Stress Intensity	11.97	64.95	5.43
Baseplate – Primary Membrane Stress Intensity	14.25	46.3	3.25
Baseplate – Local Membrane Plus Primary Bending Stress Intensity	37.29	69.45	1.86
Shell – Primary Membrane Stress Intensity	22.99	43.3	1.88



Table 3.4.9			
STRESS RESULTS FOR HI-TRAC VW WATER JACKET – ACCIDENT INTERNAL PRESSURE			
Item	Calculated Value (ksi)	Allowable Limit (ksi)	Safety Factor
Bottom Flange – Primary Membrane Plus Bending Stress	5.07	58.5	11.54
Water Jacket Shell – Primary Membrane Plus Bending Stress	7.97	58.5	7.34
Water Jacket Rib – Primary Membrane Stress	4.72	39.0	8.26
Water Jacket Shell-to- Bottom Flange Weld – Primary Shear Stress	3.70	29.4	7.94

Table 3.4.10			
STRESS RESULTS FOR HI-STORM FW LID – SNOW LOAD			
Item	Calculated Value (ksi)	Allowable Limit (ksi)	Safety Factor
Maximum Primary Membrane Stress	1.81	16.6	9.16
Maximum Primary Membrane Plus Bending Stress	1.81	24.9	13.7

Table 3.4.11

INPUT DATA USED FOR CALCULATING ANGULAR VELOCITY OF OVERPACK  
DURING NON-MECHANISTIC TIPOVER (LOAD CASE 4)

Item	Value
Maximum weight of loaded HI-STORM FW (W)	426,300 lbf <sup>†</sup>
Mid-height of maximum length HI-STORM FW (h)	119.75 in
Outer diameter of HI-STORM FW (d)	140 in
Distance between cask pivot point and cask center (r)	138.709 in
Mass moment of inertia of loaded HI-STORM FW about cask pivot point ( $I_A$ )	$1.076 \times 10^{10}$ lb-in <sup>2</sup>

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<sup>†</sup> Bounds value in Table 3.2.8.

Table 3.4.12

INPUT VALUES USED FOR CALCULATING STRESS  
IN WATER JACKET SHELL (LOAD CASE 8)

Item	Value
Mean radius of water jacket shell (R)	47.25 in
Thickness of water jacket shell (d)	0.5 in
Width of beam strip (b)	1 in
Extreme fiber distance of beam cross-section (c)	0.25 in
Unsupported span of water jacket shell ( $\theta$ )	45 deg
Distributed load on water jacket shell (w)	-75 lbf/in <sup>†</sup>
Span of distributed load on water jacket shell ( $\phi$ )	45 deg

Note: Variables are defined in Figure 3.4.35.

<sup>†</sup> Bounds accident internal pressure in Table 2.2.1 for HI-TRAC water jacket.

Table 3.4.13

Intentionally Deleted

Table 3.4.14		
KEY INPUT DATA FOR FUEL ROD INTEGRITY ANALYSIS DURING MPC REFLOOD EVENT (LOAD CASE 11)		
Item	Input Value	Source
Cladding Thickness (for reference PWR fuel), in	0.022	SAR Tables 1.0.4 and 2.1.2
Cladding OD (for reference PWR fuel), in	0.377	SAR Tables 1.0.4 and 2.1.2
Fuel Rod Pressure, psi	2,000	Ref. [3.4.24] (upper bound value)
Yield Strength of Zircaloy, psi	100,000 (at 80°F) 50,500 (at 750°F)	Ref. [3.4.21]
Tensile Strength of Zircaloy, psi	112,100 (at 80°F) 68,200 (at 750°F)	Ref. [3.4.21]
Elastic Modulus of Zircaloy, $\times 10^6$ psi	13.42 (at 80°F) 10.4 (at 750°F)	Ref. [3.4.21]
Coefficient of Thermal Expansion of Zircaloy, $\times 10^{-6}$ in/in/°F	3.3 (at 80°F) 4.5 (at 750°F)	Ref. [3.4.22]
Poisson's Ratio of Zircaloy	0.4	Appendix C of Ref. [3.4.23]

Table 3.4.15	
MAXIMUM RESULTS FOR FUEL ROD INTEGRITY ANALYSIS DURING MPC REFLOOD EVENT (LOAD CASE 11)	
Result	Value
Maximum Stress in Fuel Rod Cladding	29,995 psi
Maximum Strain in Fuel Rod Cladding	$2.66 \times 10^{-3}$

Table 3.4.16			
CASK SLIDING DISPLACEMENTS DUE LARGE MISSILE IMPACT (LOAD CASE 3)			
Cask	Calculated Sliding Displacement (ft)	Allowable Sliding Displacement (ft)	Safety Factor
HI-STORM FW	0.454	3.33 (cask to cask)	7.33
		6.2 (cask to edge of ISFSI pad)	13.6
HI-TRAC VW	1.23	None Established	-



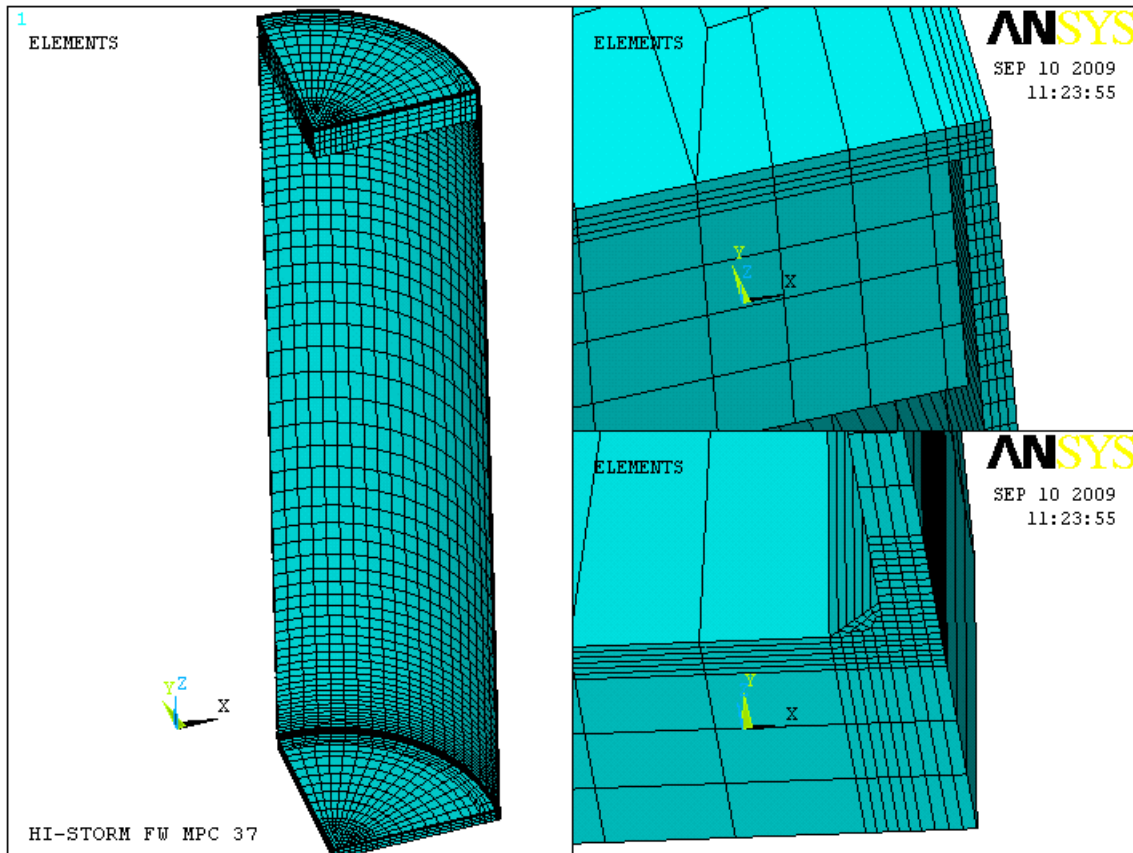


Figure 3.4.1: ANSYS Model of MPC Enclosure Vessel – Normal Handling

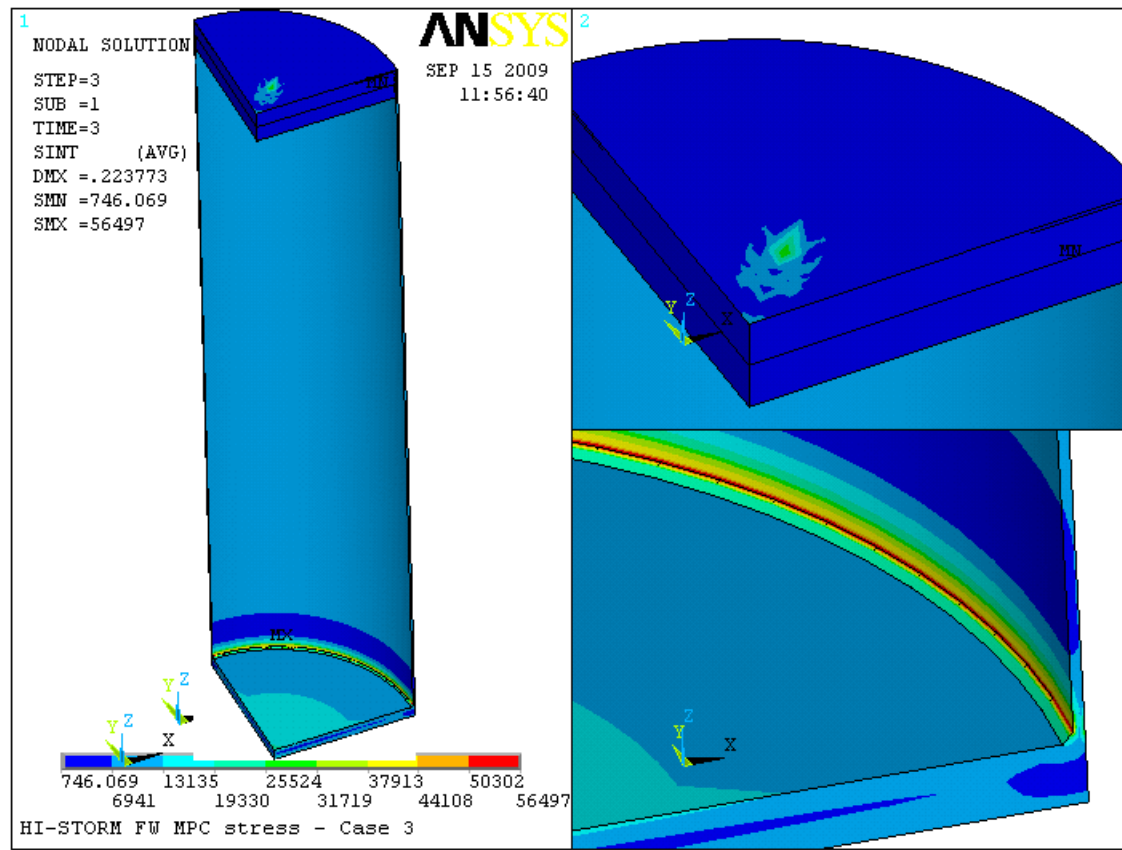


Figure 3.4.2: Stress Intensity Distribution in MPC Enclosure Vessel – Normal Handling

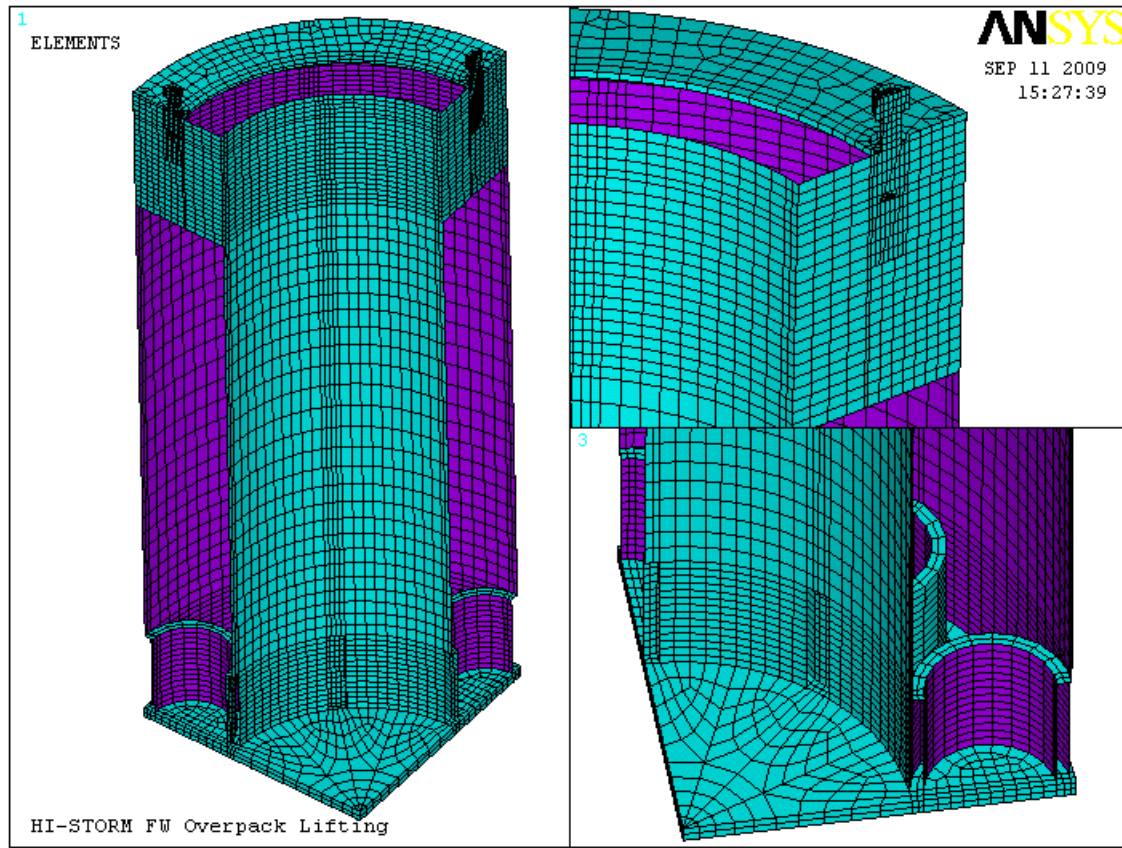


Figure 3.4.3: ANSYS Model of HI-STORM FW Overpack – Normal Handling

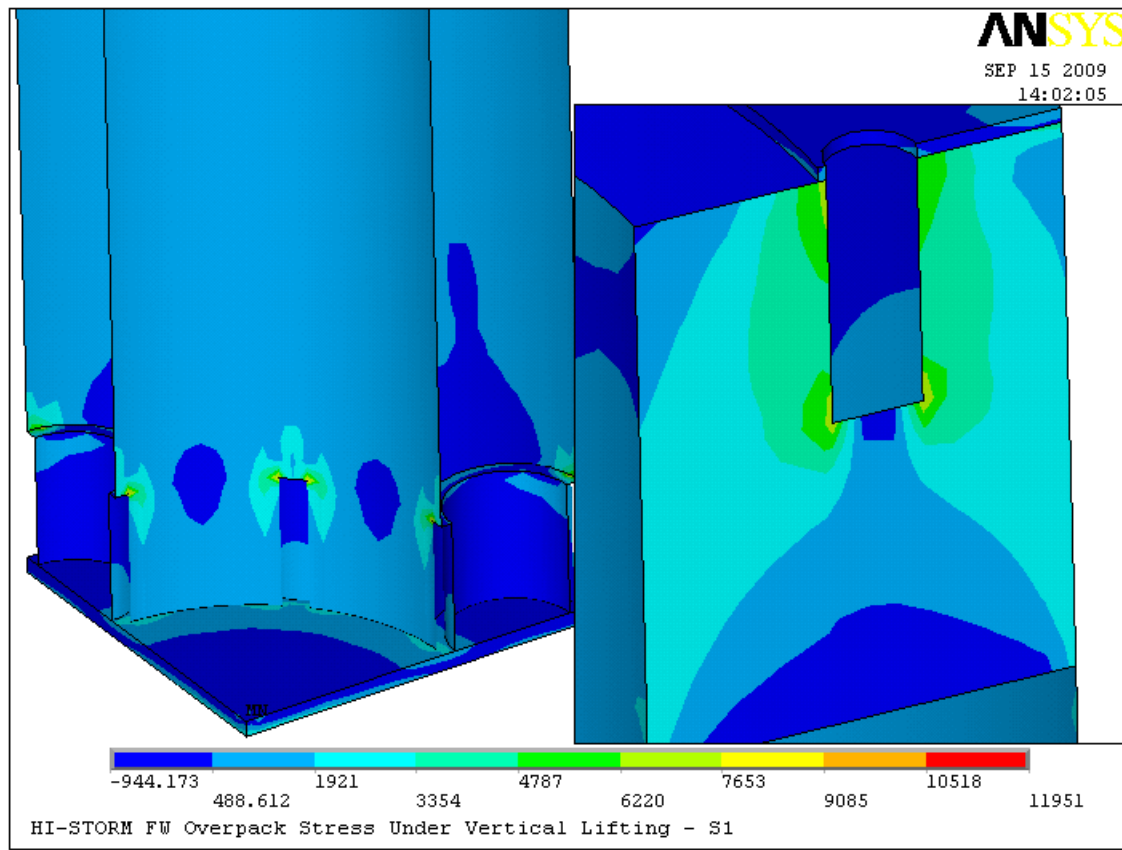


Figure 3.4.4: Stress Distribution in HI-STORM FW Overpack – Normal Handling

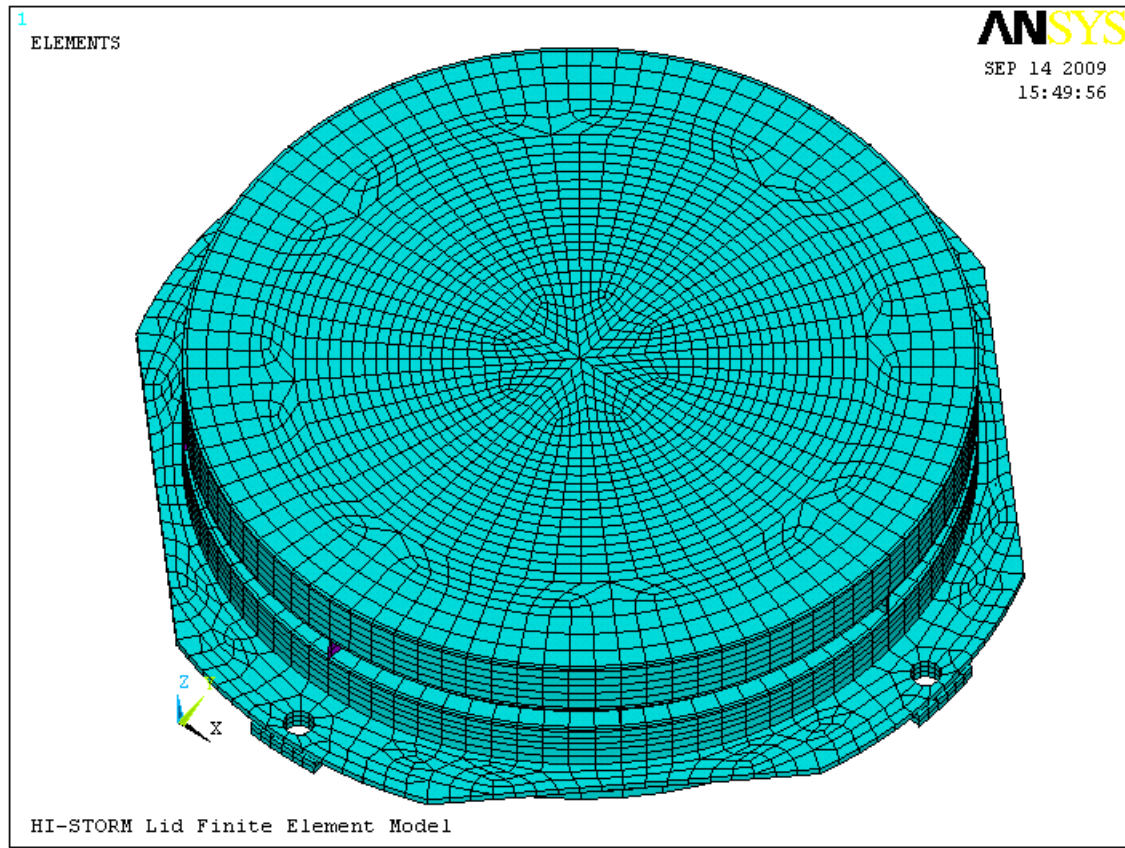


Figure 3.4.5: ANSYS Model of HI-STORM FW Lid – Normal Handling

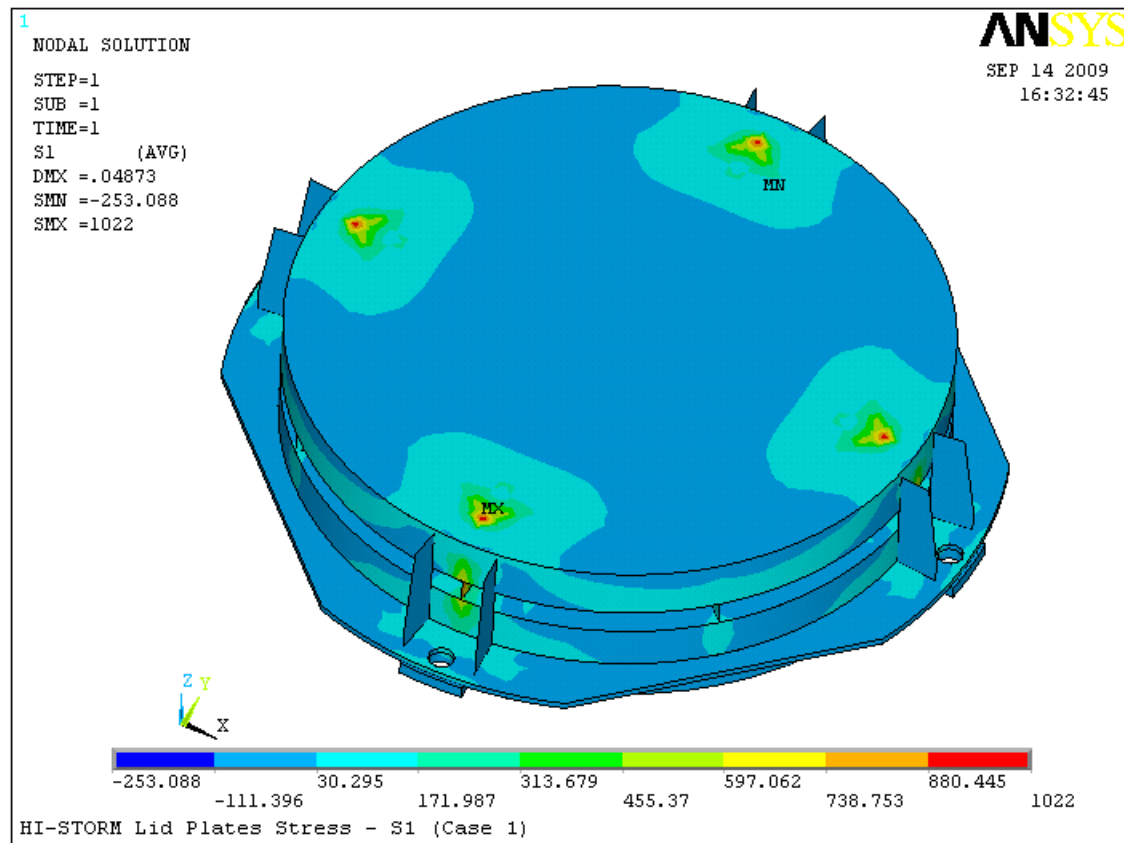


Figure 3.4.6: Stress Distribution in HI-STORM FW Lid – Normal Handling

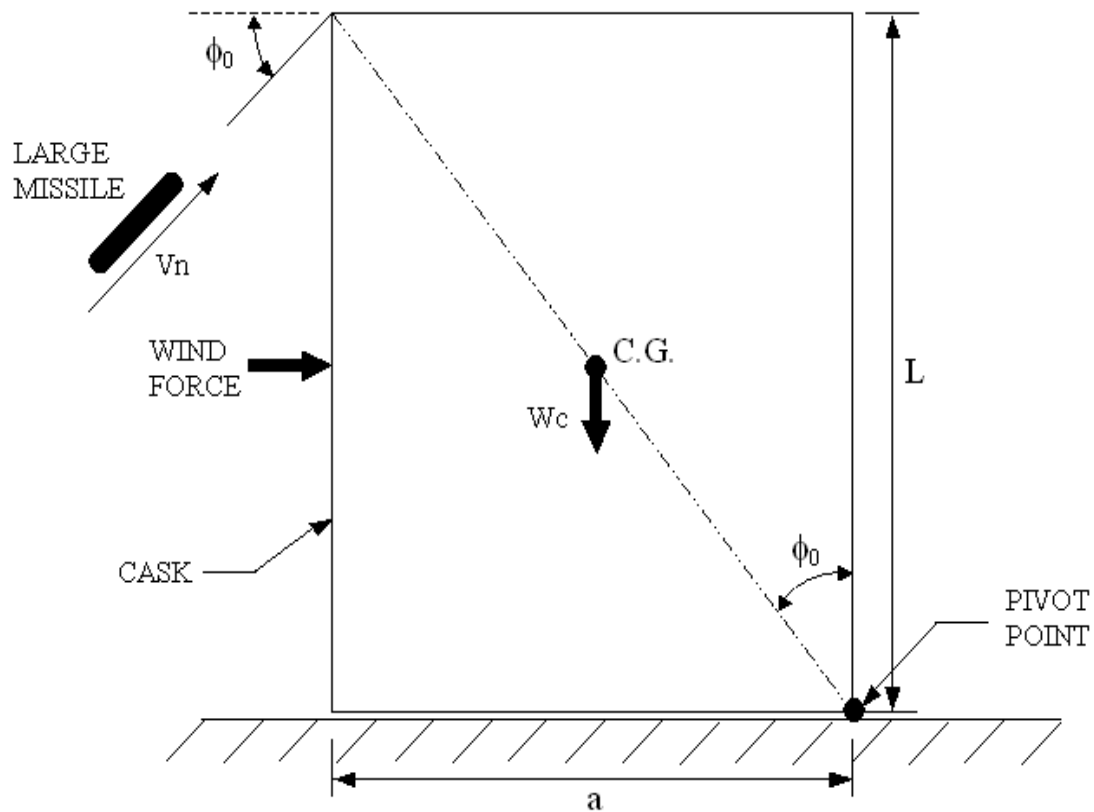


Figure 3.4.7: Free Body Diagram of Cask for Large Missile Strike/Tornado Event

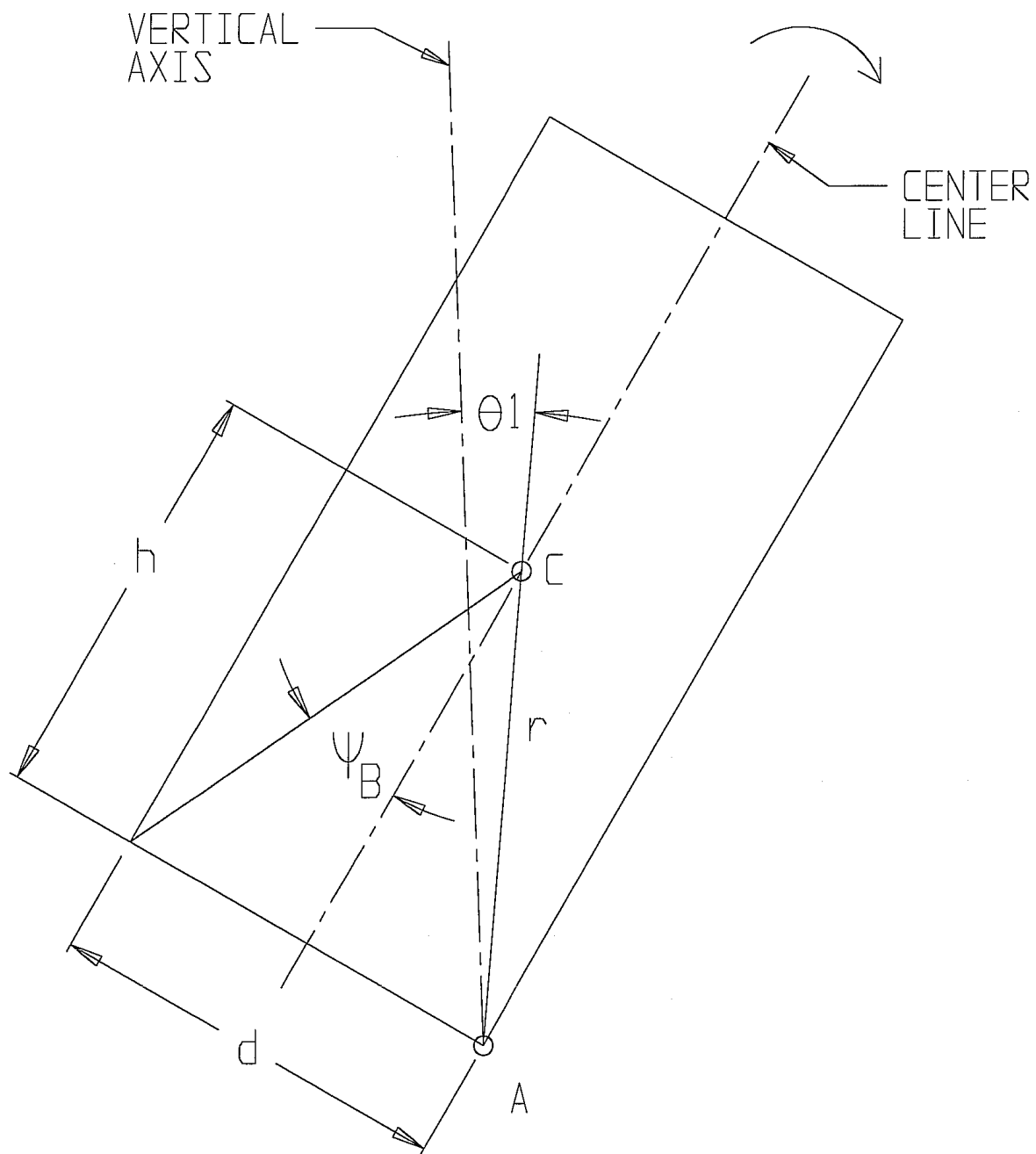


Figure 3.4.8: Cask Configuration at Incipient Tipping



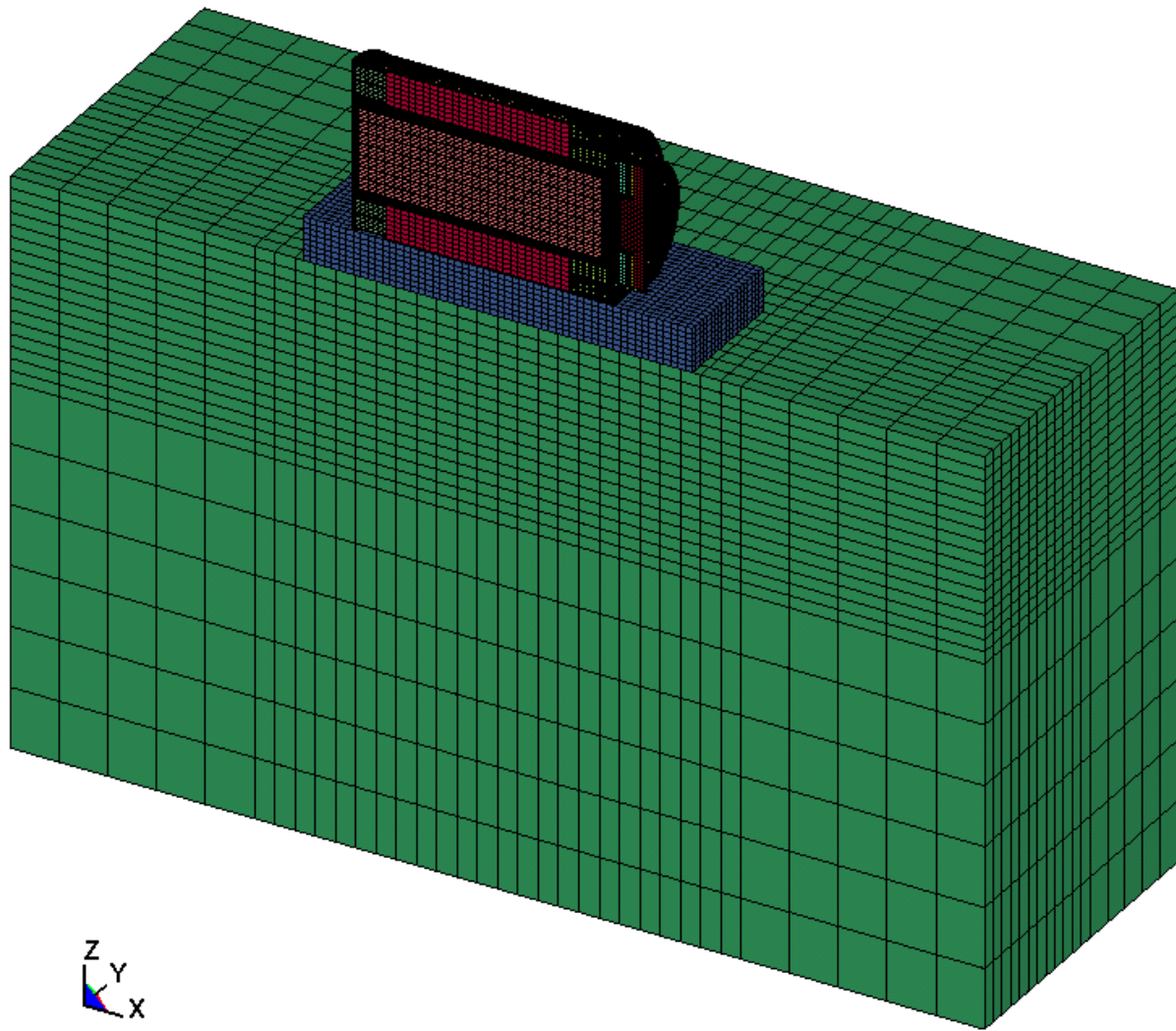
**HISTORM FW (loaded with MPC 37) TIPOVER**

Figure 3.4.9A: LS-DYNA Tipover Model – HI-STORM FW Loaded with MPC-37

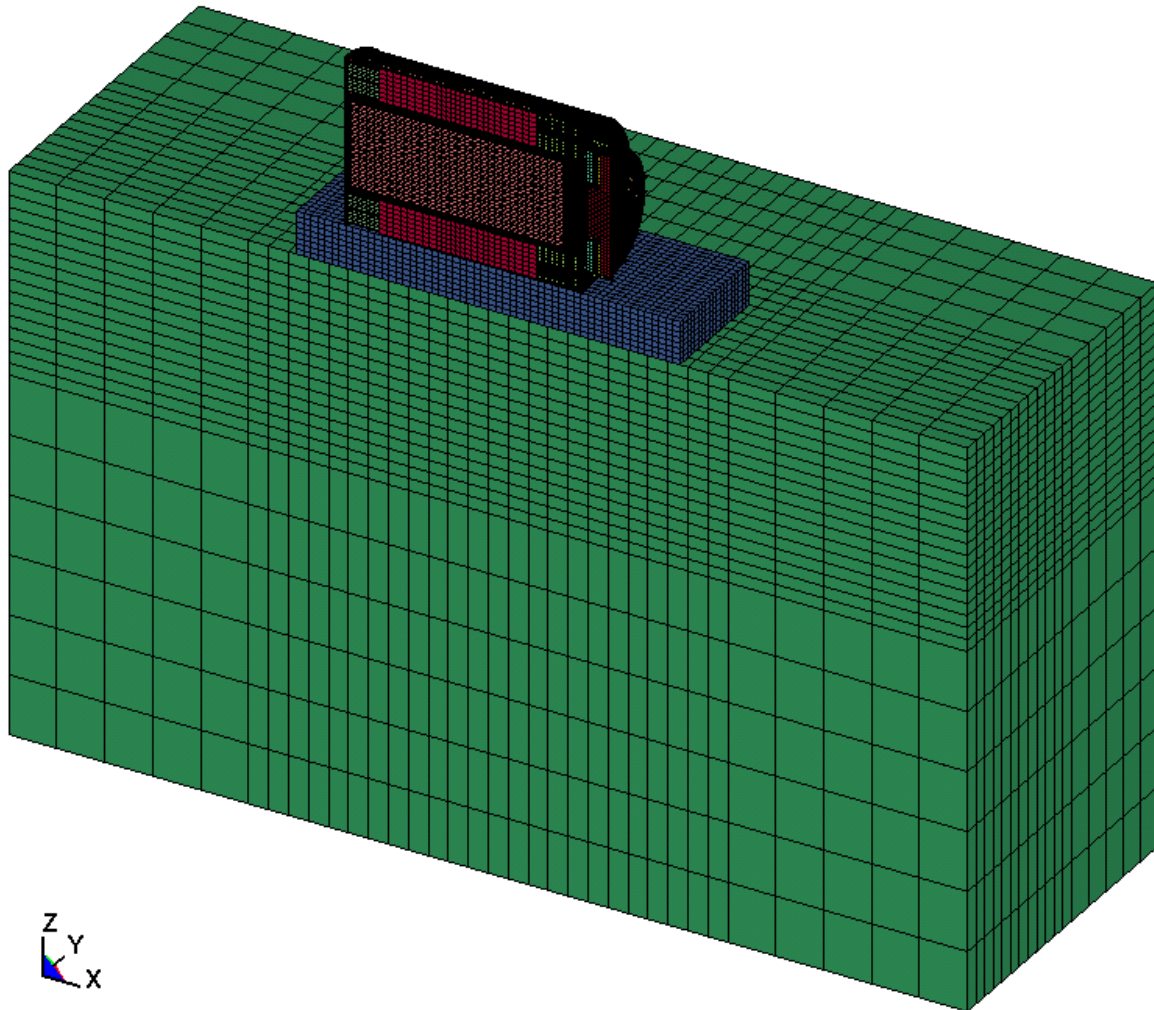
**HISTORM FW (loaded with MPC 89) TIPOVER**

Figure 3.4.9B: LS-DYNA Tipover Model – HI-STORM FW Loaded with MPC-89

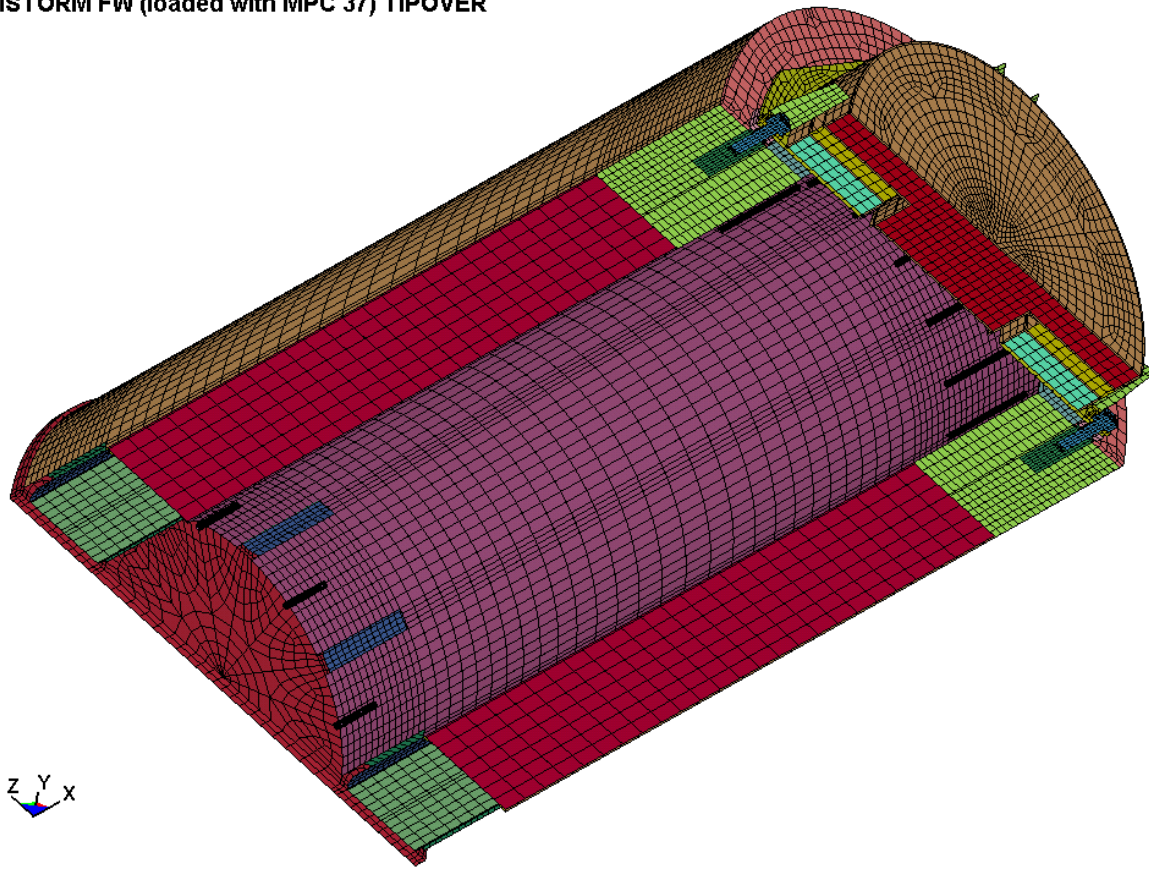
**HISTORM FW (loaded with MPC 37) TIPOVER**

Figure 3.4.10A: LS-DYNA Model – HI-STORM FW for MPC-37



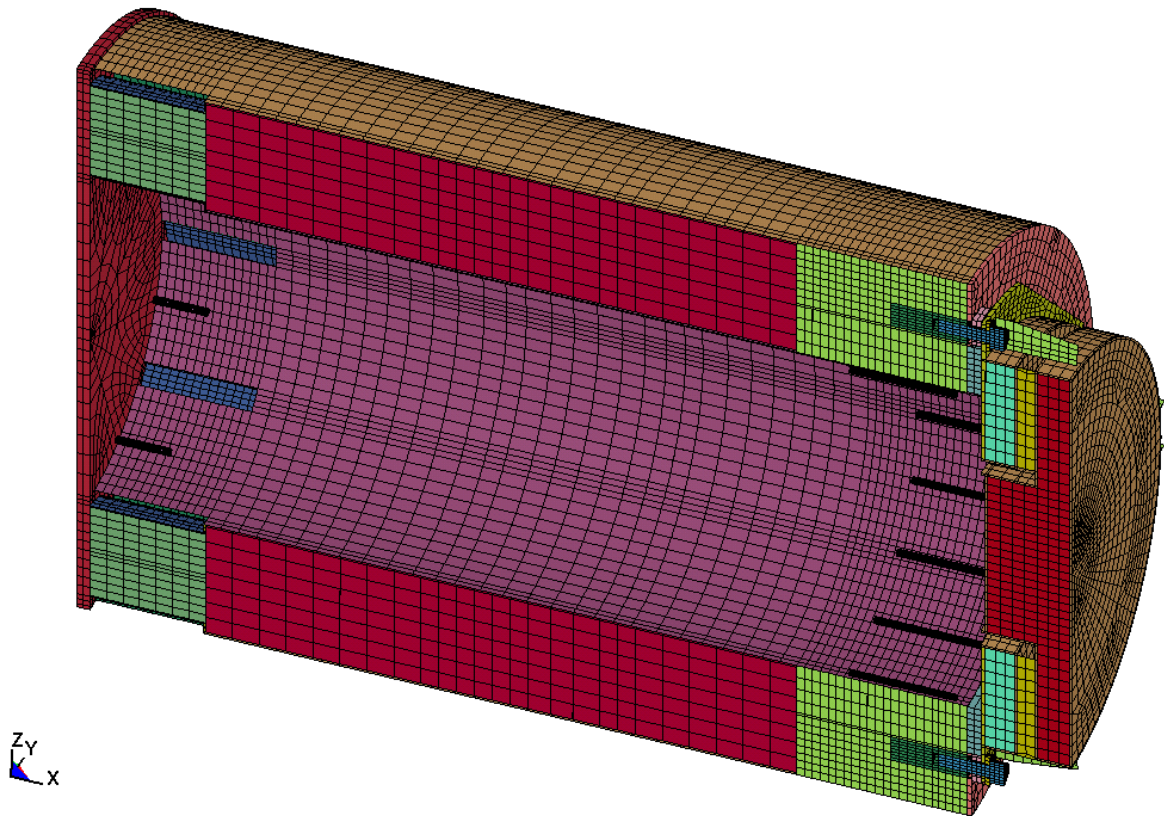
**HISTORM FW (loaded with MPC 89) TIPOVER**

Figure 3.4.10B: LS-DYNA Model – HI-STORM FW for MPC-89

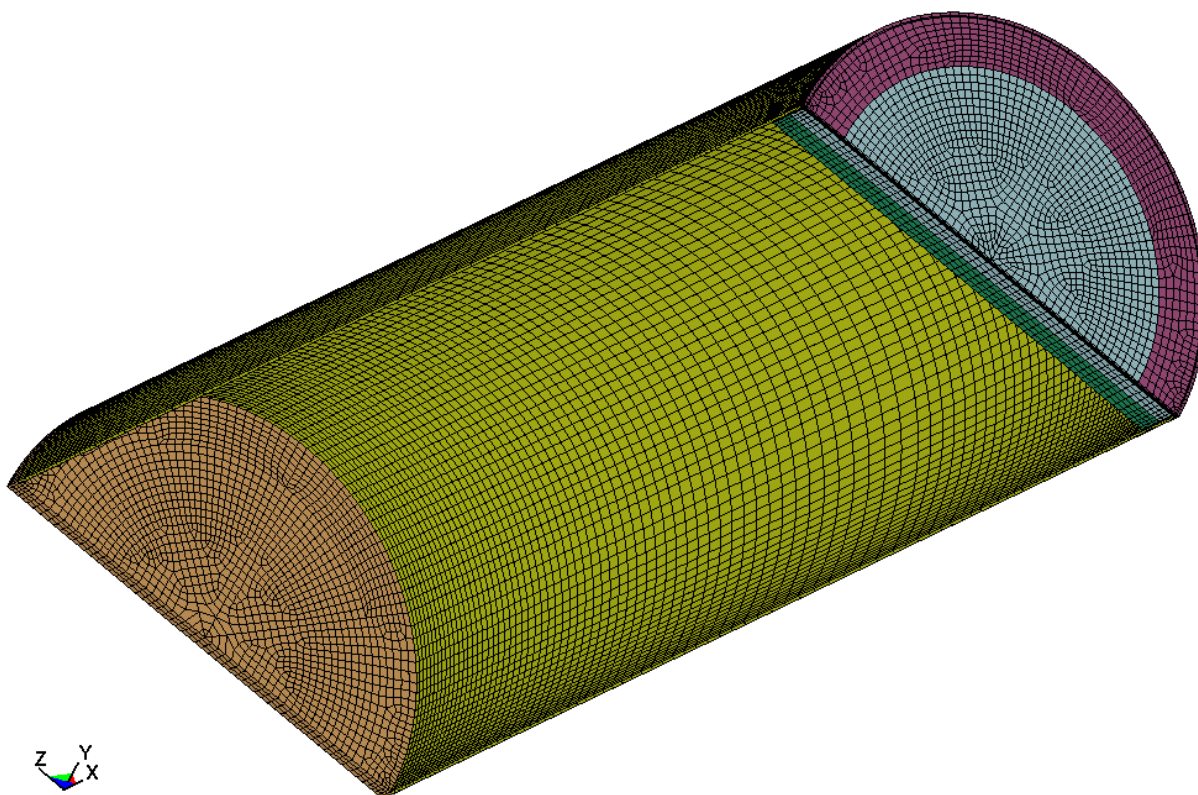
**HISTORM FW (loaded with MPC 37) TIPOVER**

Figure 3.4.11A: LS-DYNA Model – MPC-37 Enclosure Vessel

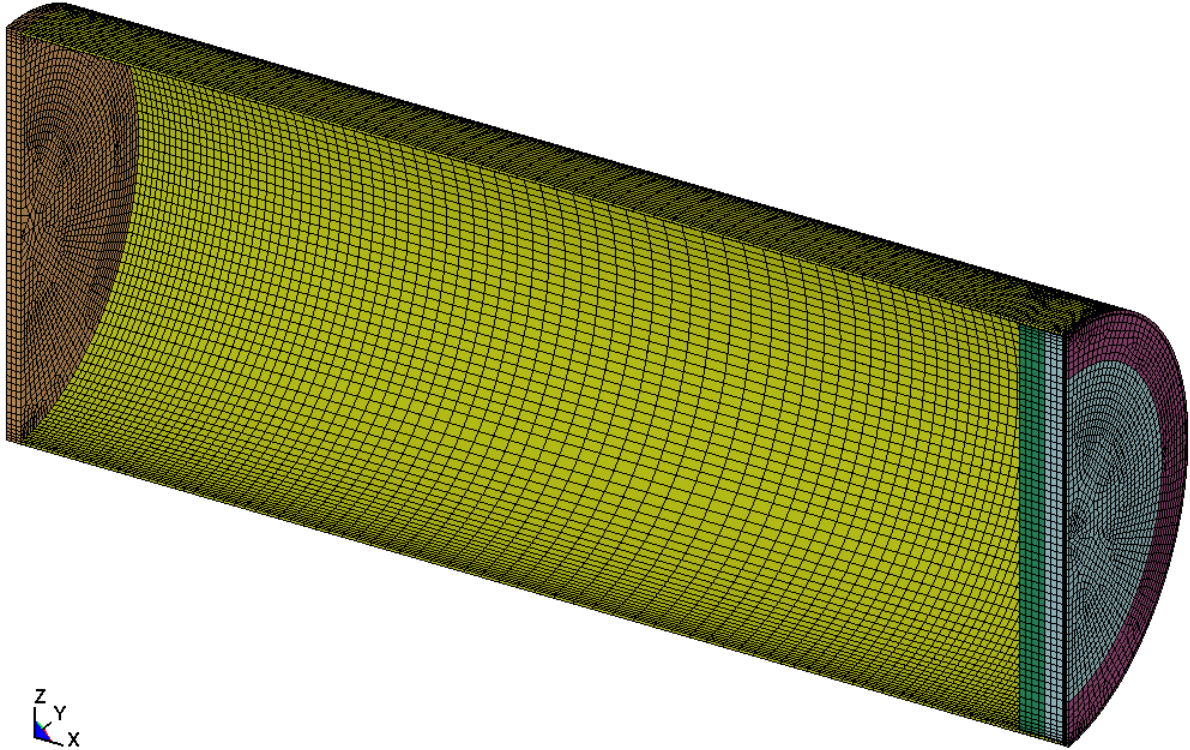
**HISTORM FW (loaded with MPC 89) TIPOVER**

Figure 3.4.11B: LS-DYNA Model – MPC-89 Enclosure Vessel



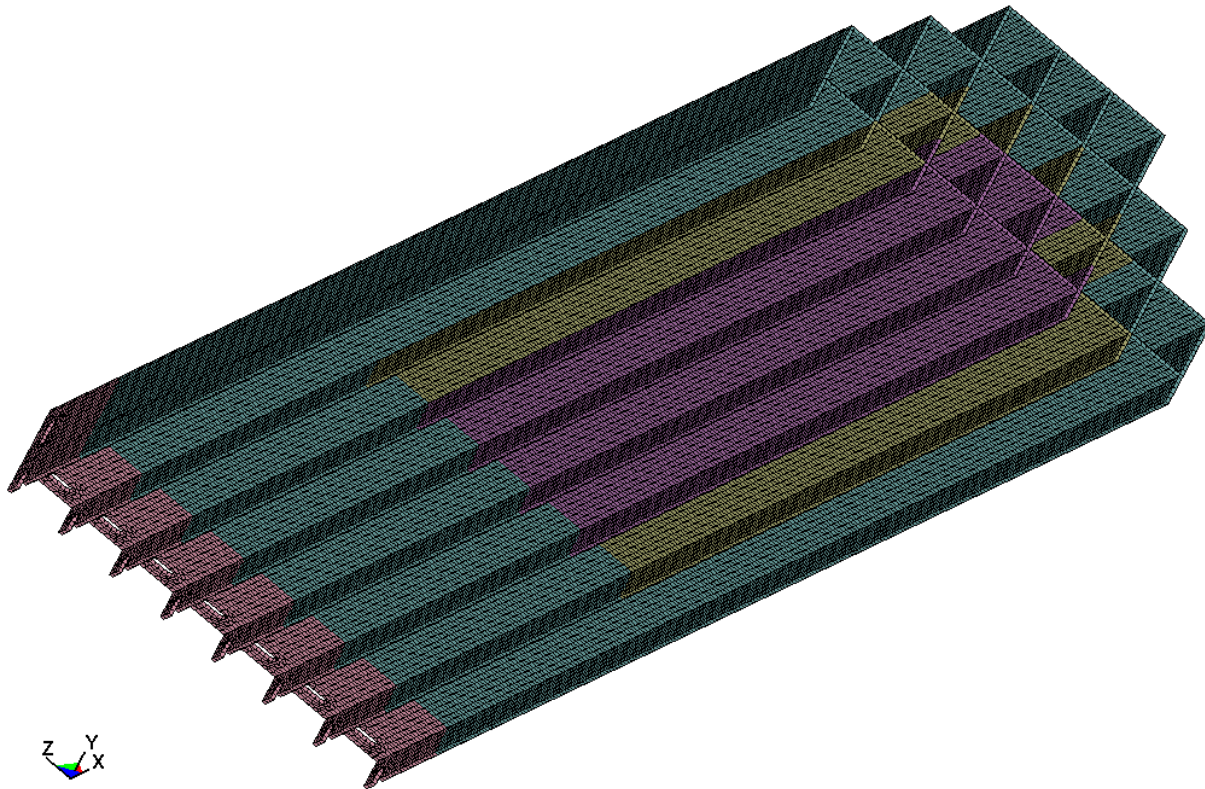
**HISTORM FW (loaded with MPC 37) TIPOVER**

Figure 3.4.12A: LS-DYNA Model – MPC-37 Fuel Basket  
(note: the different colors represent regions with bounding temperatures of 340°C, 325°C, 300°C and 250°C, respectively)

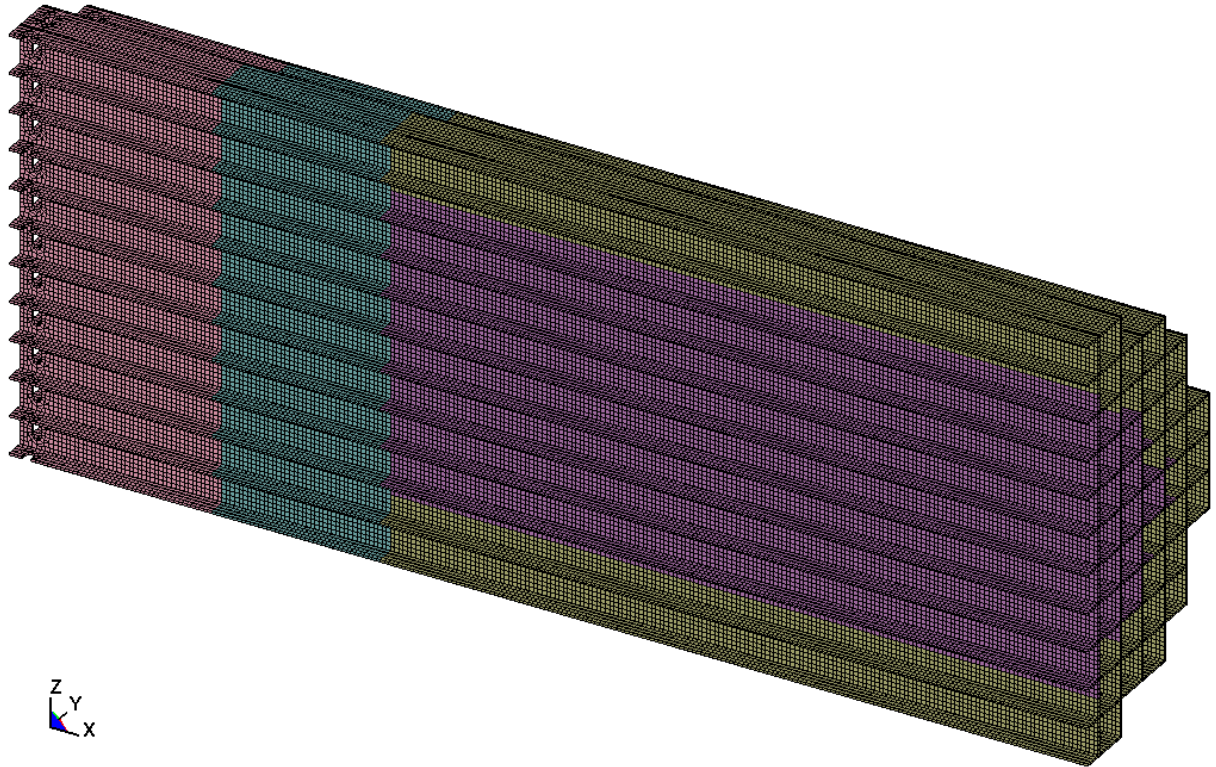
**HISTORM FW (loaded with MPC 89) TIPOVER**

Figure 3.4.12B: LS-DYNA Model – MPC-89 Fuel Basket  
(note: the different colors represent regions with bounding temperatures of 325°C, 300°C, 250°C and 200°C, respectively)



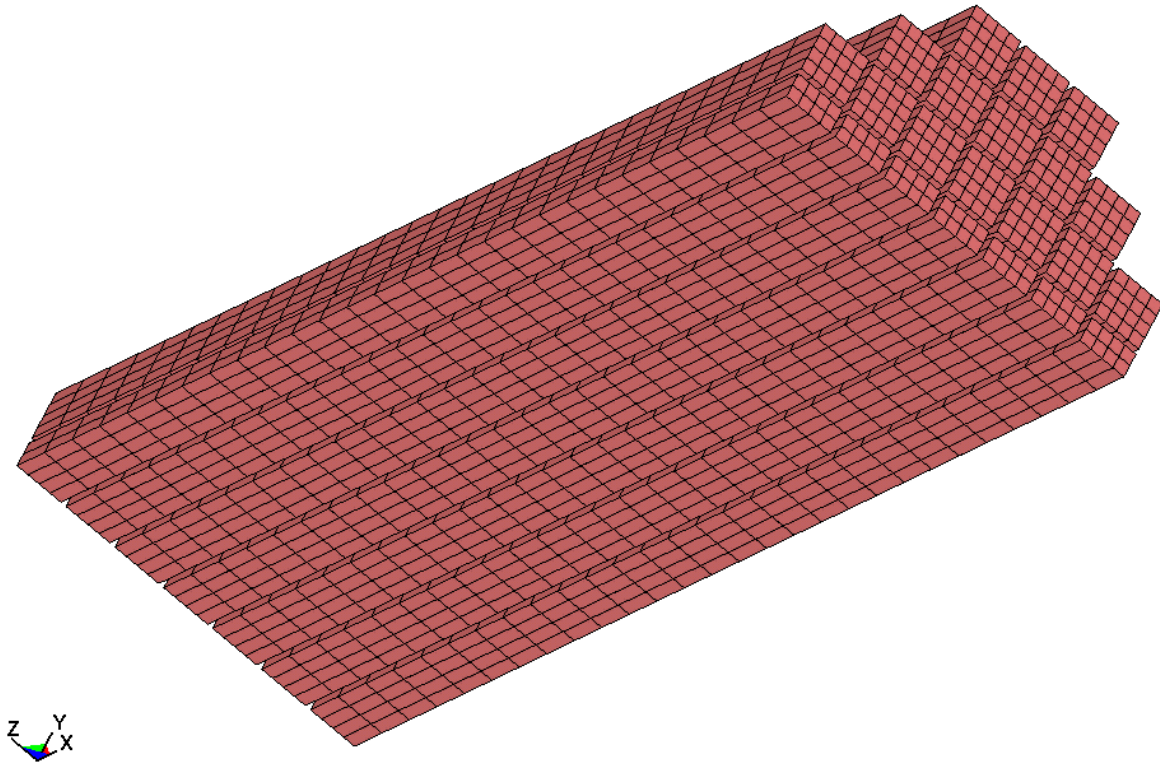
**HISTORM FW (loaded with MPC 37) TIPOVER**

Figure 3.4.13A: LS-DYNA Model – PWR Fuel Assemblies

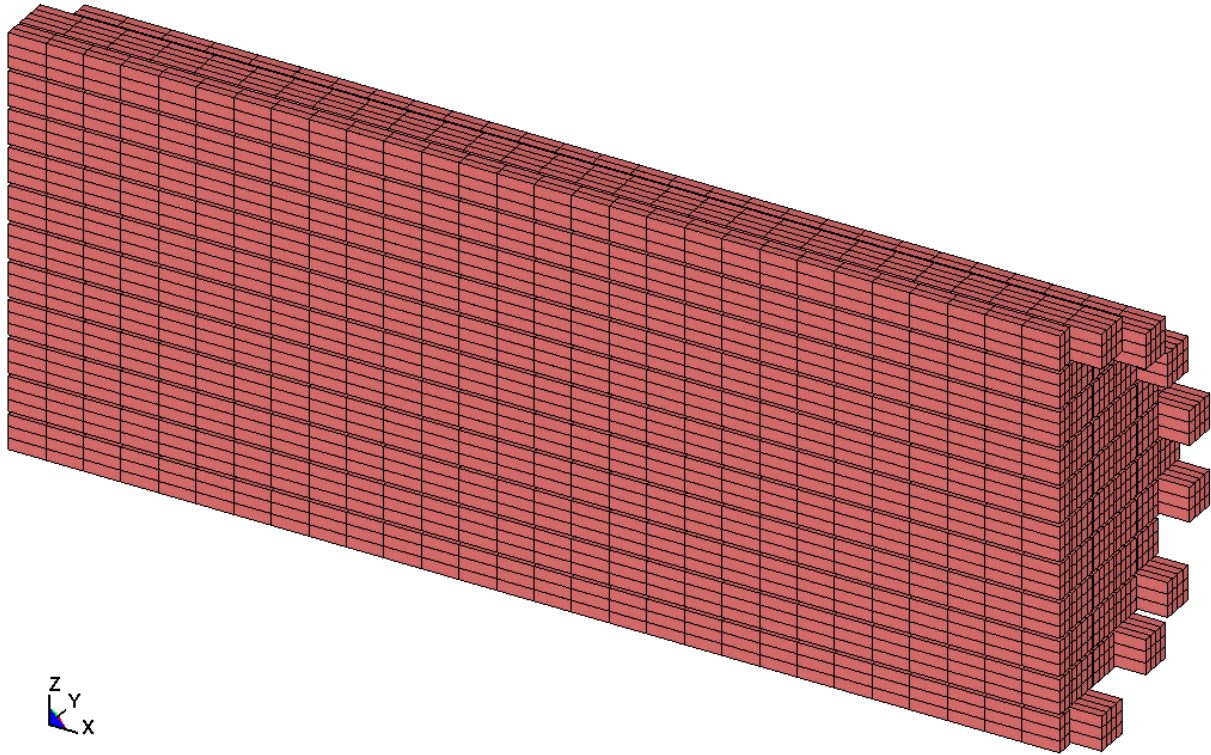
**HISTORM FW (loaded with MPC 89) TIPOVER**

Figure 3.4.13B: LS-DYNA Model – BWR Fuel Assemblies & Damaged Fuel Containers

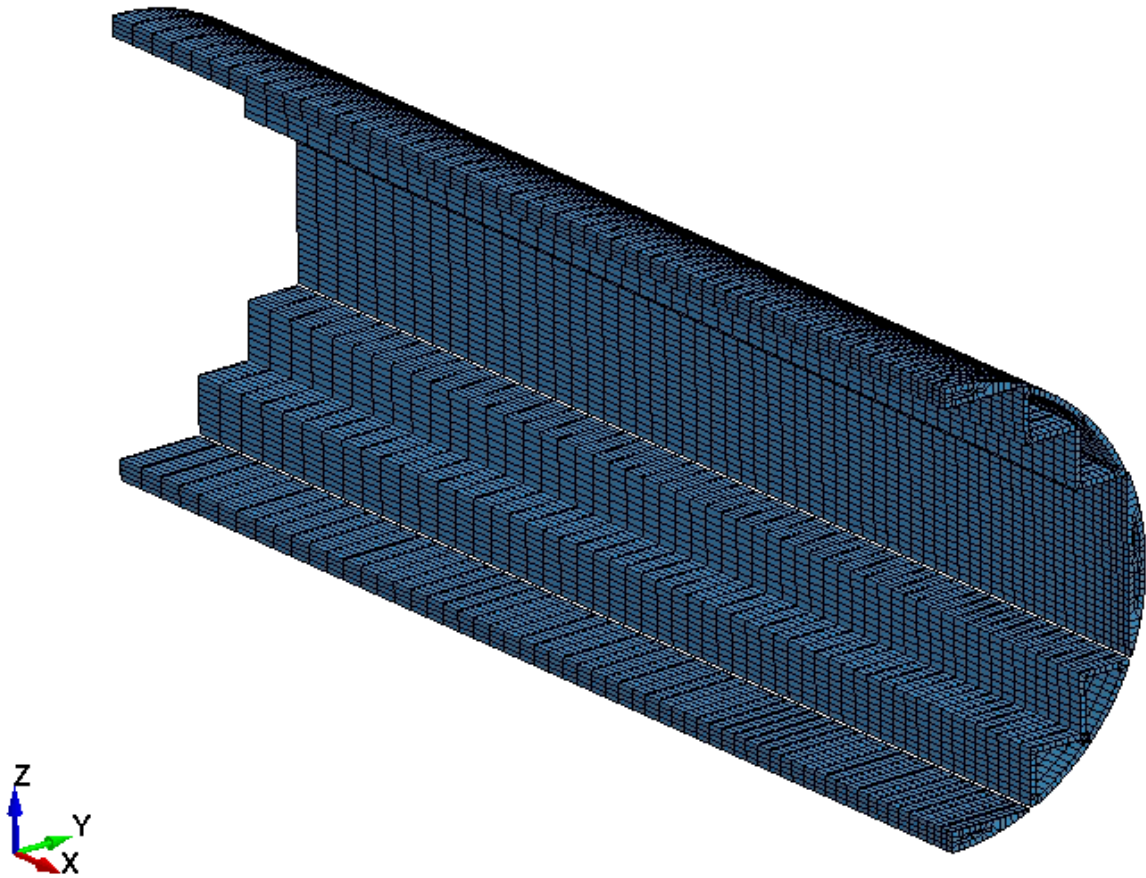


Figure 3.4.14A: LS-DYNA Model – MPC-37 Fuel Basket Shims

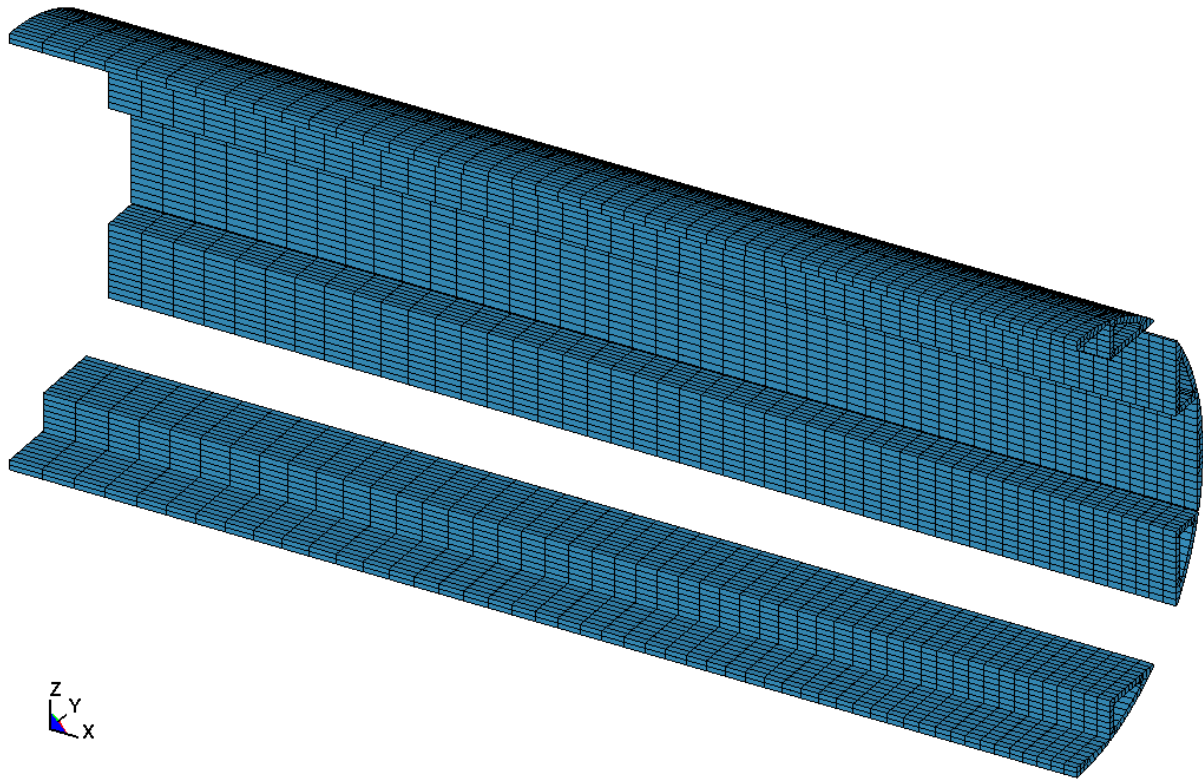
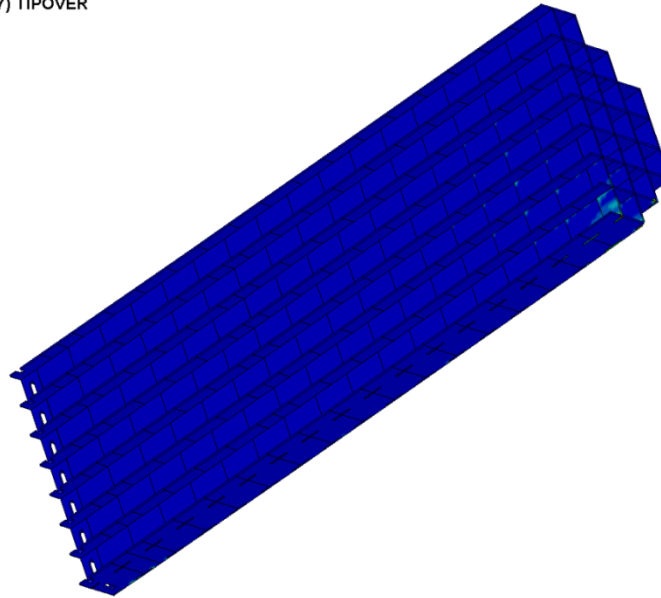
**HISTORM FW (loaded with MPC 89) TIPOVER**

Figure 3.4.14B: LS-DYNA Model – MPC-89 Fuel Basket Shims



HISTORM FW (loaded with MPC 37) TIPOVER  
 Time = 0.050001  
 Contours of Effective Plastic Strain  
 max ipt. value  
 min=0, at elem# 524233  
 max=0.125121, at elem# 1000366

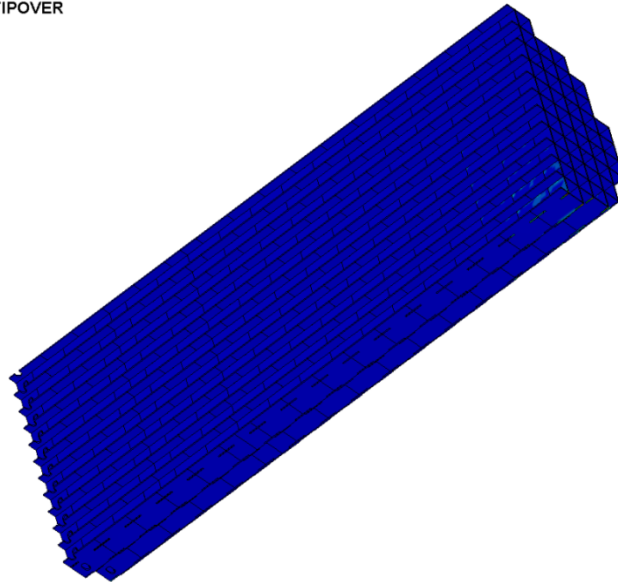


Fringe Levels

1.251e-01
1.126e-01
1.001e-01
8.759e-02
7.507e-02
6.256e-02
5.005e-02
3.754e-02
2.502e-02
1.251e-02
0.000e+00

Figure 3.4.15A: Maximum Plastic Strain – MPC-37 Fuel Basket

HISTORM FW (loaded with MPC 89) TIPOVER  
 Time = 0.05  
 Contours of Effective Plastic Strain  
 max lpt. value  
 min=0, at elem# 537641  
 max=0.131194, at elem# 606300



Fringe Levels

1.312e-01
1.181e-01
1.050e-01
9.184e-02
7.872e-02
6.560e-02
5.248e-02
3.936e-02
2.624e-02
1.312e-02
0.000e+00

Figure 3.4.15B: Maximum Plastic Strain – MPC-89 Fuel Basket

**HISTORM FW (loaded with MPC 37) TIPOVE**

Time = 0.05

Contours of Effective Plastic Strain

max ipt. value

min=0, at elem# 400433

max=0.0753262, at elem# 424290

**Fringe Levels**

7.533e-02

6.779e-02

6.026e-02

5.273e-02

4.520e-02

3.766e-02

3.013e-02

2.260e-02

1.507e-02

7.533e-03

0.000e+00

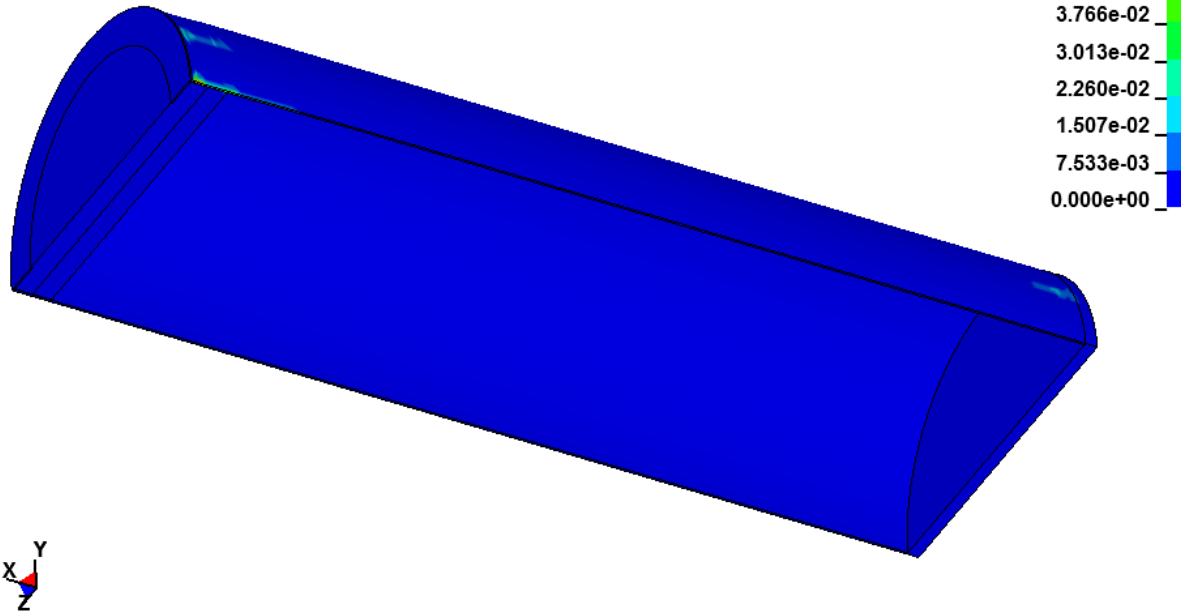


Figure 3.4.16A: Maximum Plastic Strain – MPC-37 Enclosure Vessel

**HISTORM FW (loaded with MPC 89) TIPOVE**

Time = 0.05

Contours of Effective Plastic Strain

max ipt. value

min=0, at elem# 400433

max=0.095544, at elem# 424218

**Fringe Levels**

9.554e-02

8.599e-02

7.644e-02

6.688e-02

5.733e-02

4.777e-02

3.822e-02

2.866e-02

1.911e-02

9.554e-03

0.000e+00

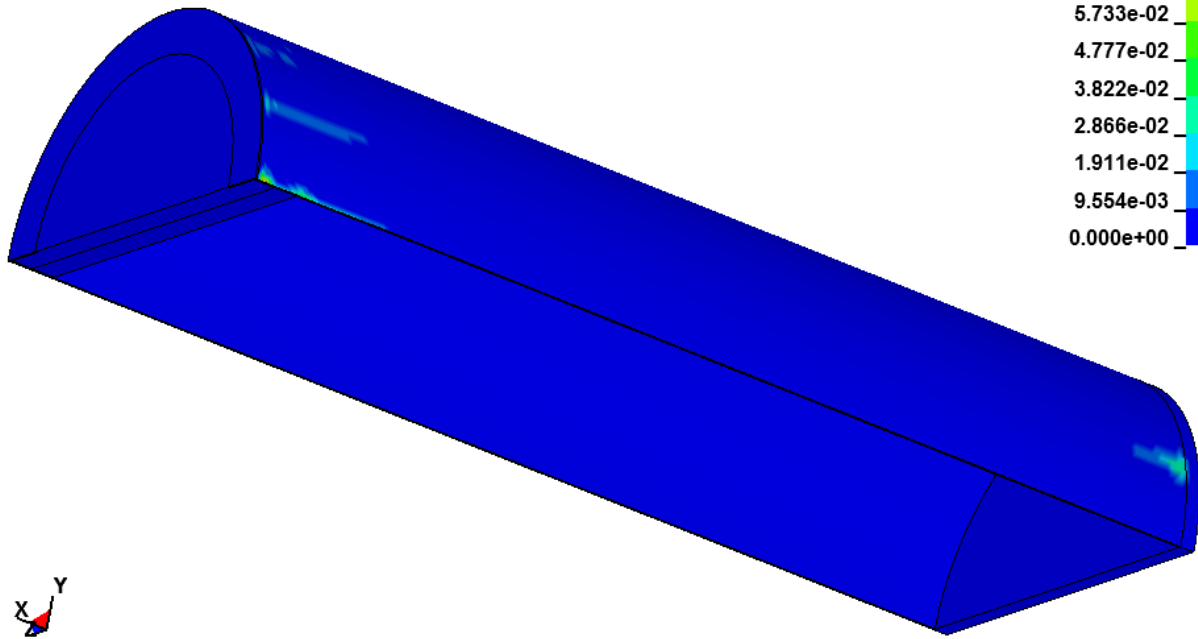


Figure 3.4.16B: Maximum Plastic Strain – MPC-89 Enclosure Vessel



**HISTORM FW (loaded with MPC 37) TIPOVE**

Time = 0.05

Contours of Effective Plastic Strain

max ipt. value

min=0, at elem# 43717

max=0.128675, at elem# 20165

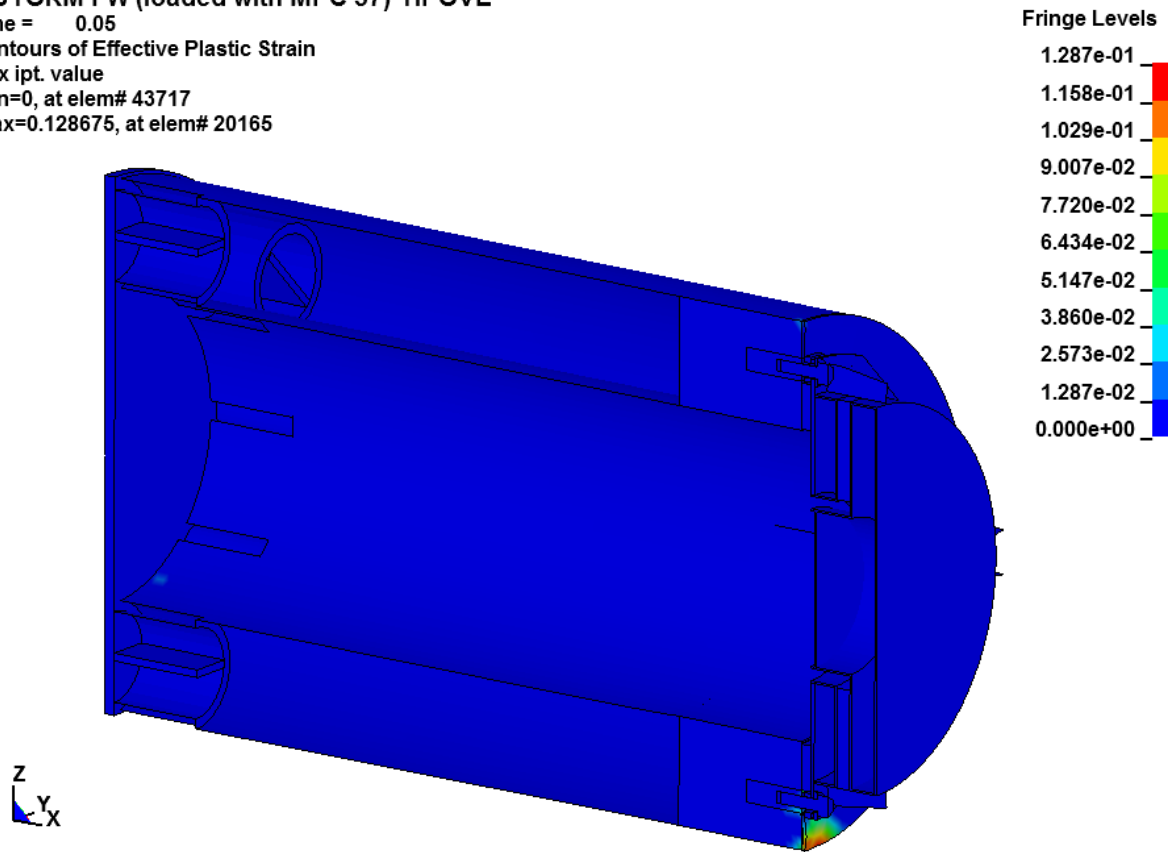


Figure 3.4.17A: Maximum Plastic Strain – HI-STORM FW Overpack  
(for MPC-37, Excluding MPC Guide Tubes)

**HISTORM FW (loaded with MPC 89) TIPOVE**

Time = 0.05

Contours of Effective Plastic Strain

max ipt. value

min=0, at elem# 43717

max=0.150564, at elem# 20165

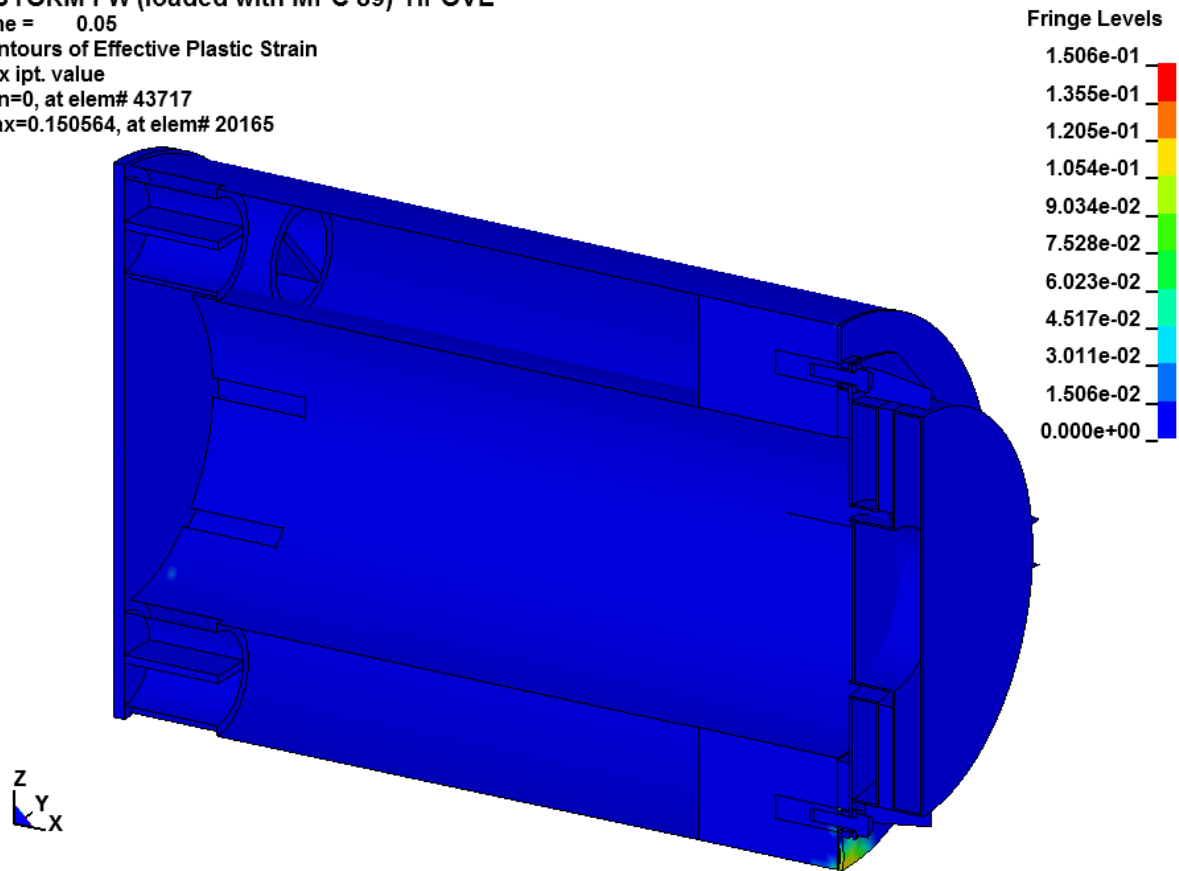


Figure 3.4.17B: Maximum Plastic Strain – HI-STORM FW Overpack  
(for MPC-89, Excluding MPC Guide Tubes)

**HISTORM FW (loaded with MPC 37) TIPOVE**

Time = 0.05

Contours of Effective Plastic Strain

max ipt. value

min=0, at elem# 43717

max=0.0065258, at elem# 44005


**Fringe Levels**

6.526e-03

5.873e-03

5.221e-03

4.568e-03

3.915e-03

3.263e-03

2.610e-03

1.958e-03

1.305e-03

6.526e-04

0.000e+00

Figure 3.4.18A: Maximum Plastic Strain –  
HI-STORM FW Overpack (for MPC-37) Closure Lid Bolts

**HISTORM FW (loaded with MPC 89) TIPOVE**

Time = 0.05

Contours of Effective Plastic Strain

max ipt. value

min=0, at elem# 43717

max=0.00423532, at elem# 44034

**Fringe Levels**

4.235e-03

3.812e-03

3.388e-03

2.965e-03

2.541e-03

2.118e-03

1.694e-03

1.271e-03

8.471e-04

4.235e-04

0.000e+00



Figure 3.4.18B: Maximum Plastic Strain –  
HI-STORM FW Overpack (for MPC-37) Closure Lid Bolts

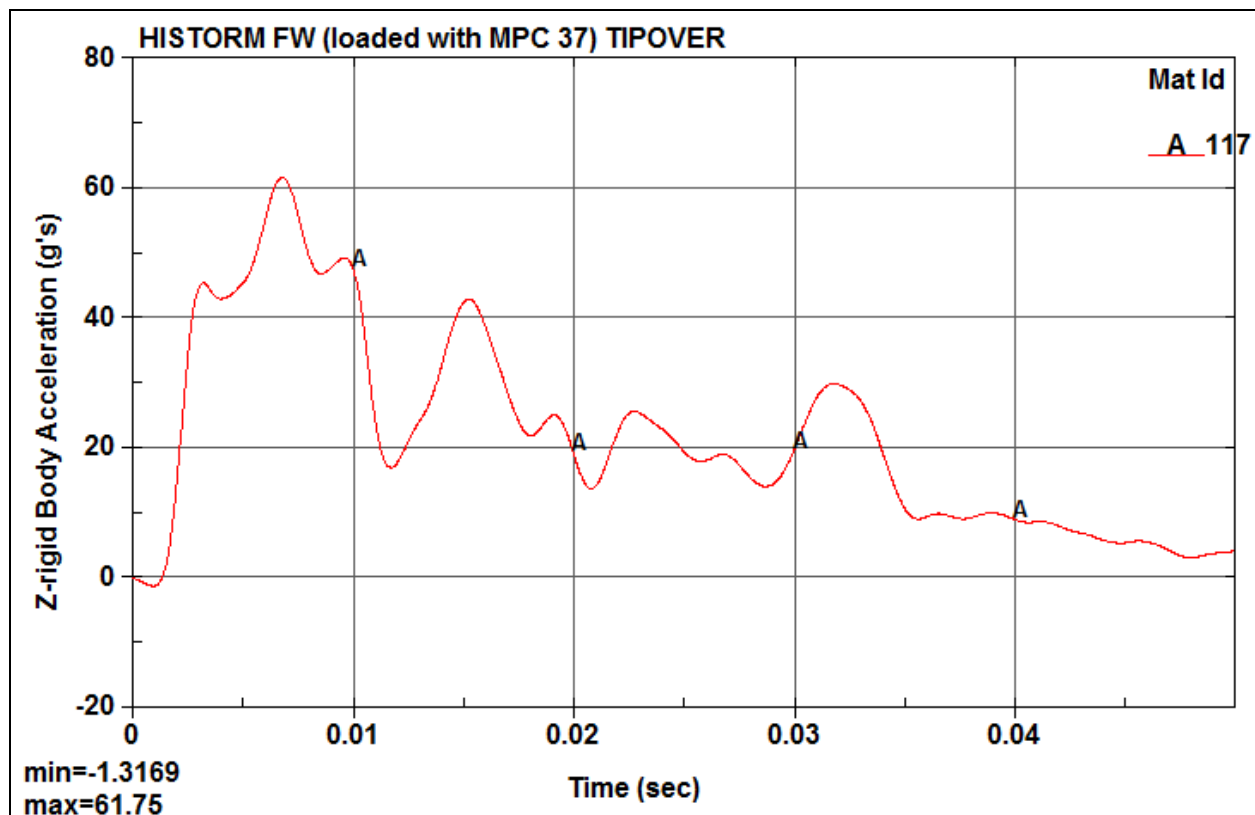


Figure 3.4.19A: Vertical Rigid Body Deceleration Time History –  
Cask Lid Concrete (for HI-STORM FW Loaded with MPC-37)

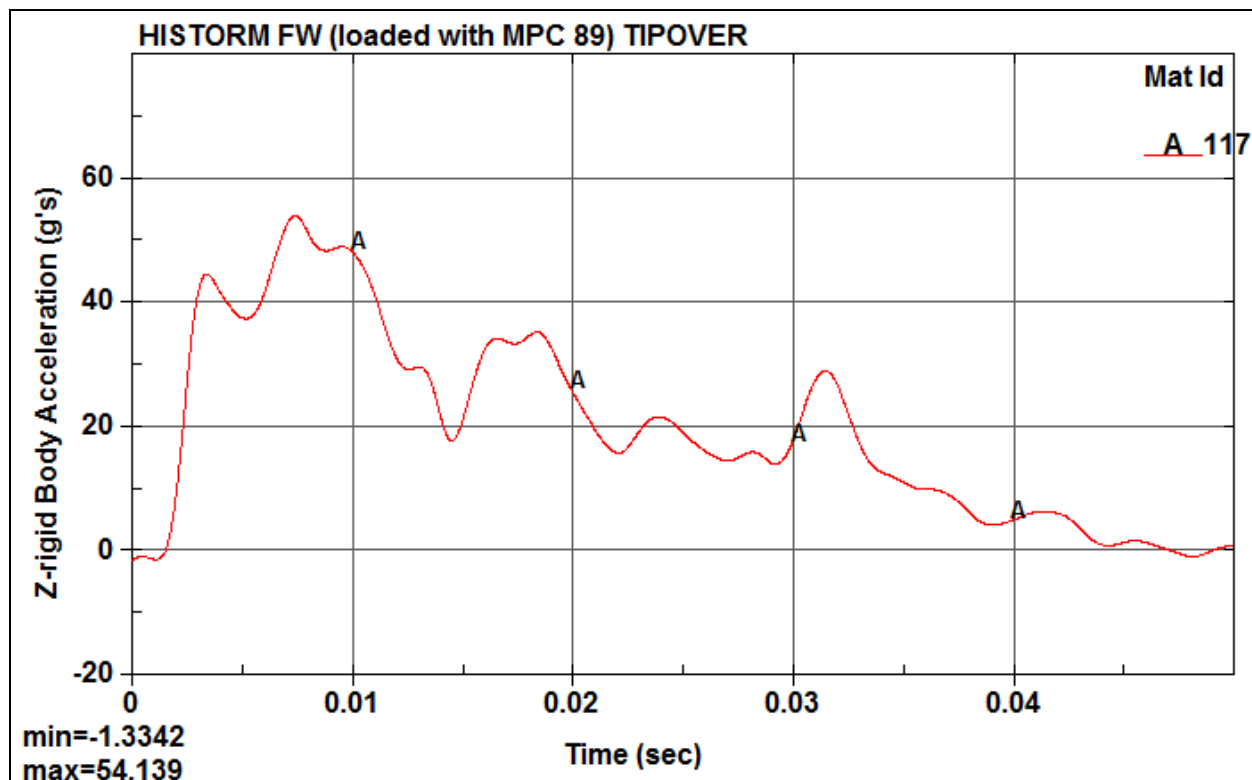


Figure 3.4.19B: Vertical Rigid Body Deceleration Time History –  
Cask Lid Concrete (for HI-STORM FW Loaded with MPC-89)

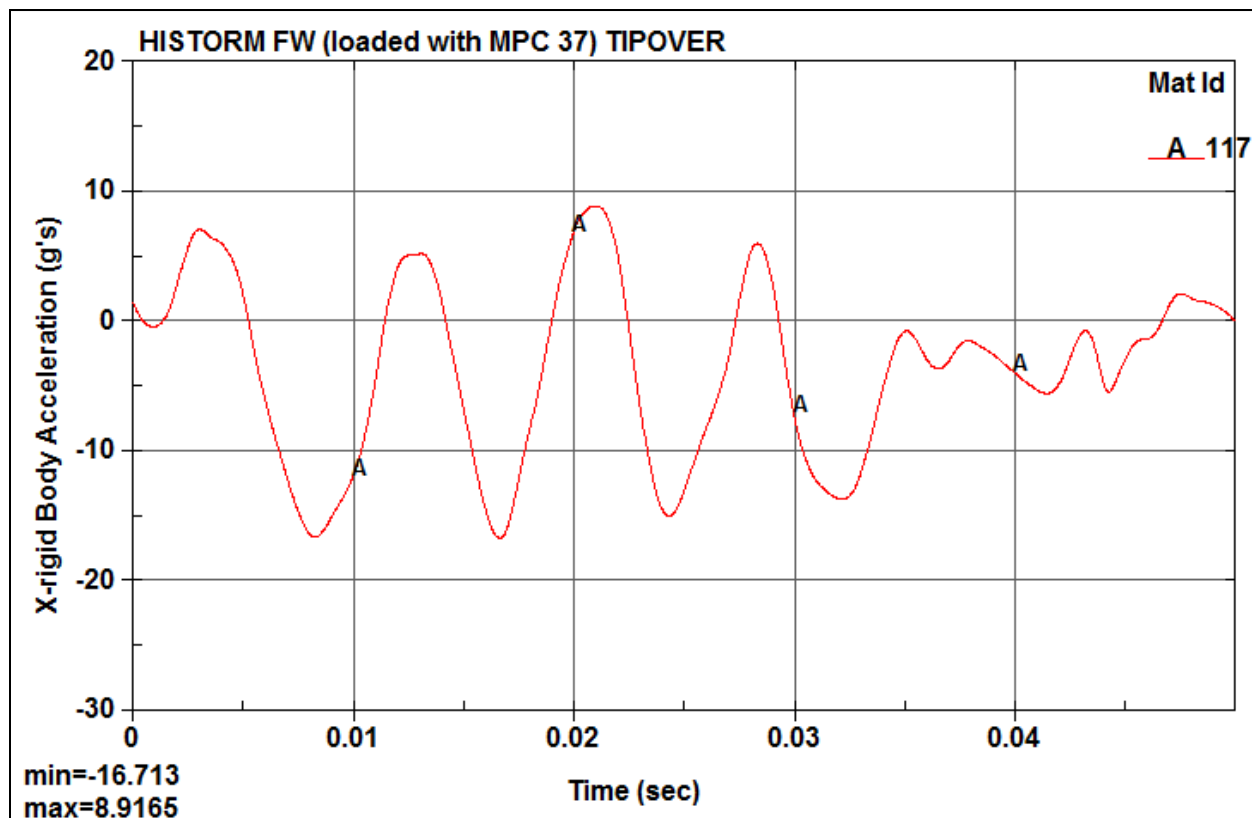


Figure 3.4.20A: Horizontal Rigid Body Deceleration Time History –  
Cask Lid Concrete (for HI-STORM FW Loaded with MPC-37)

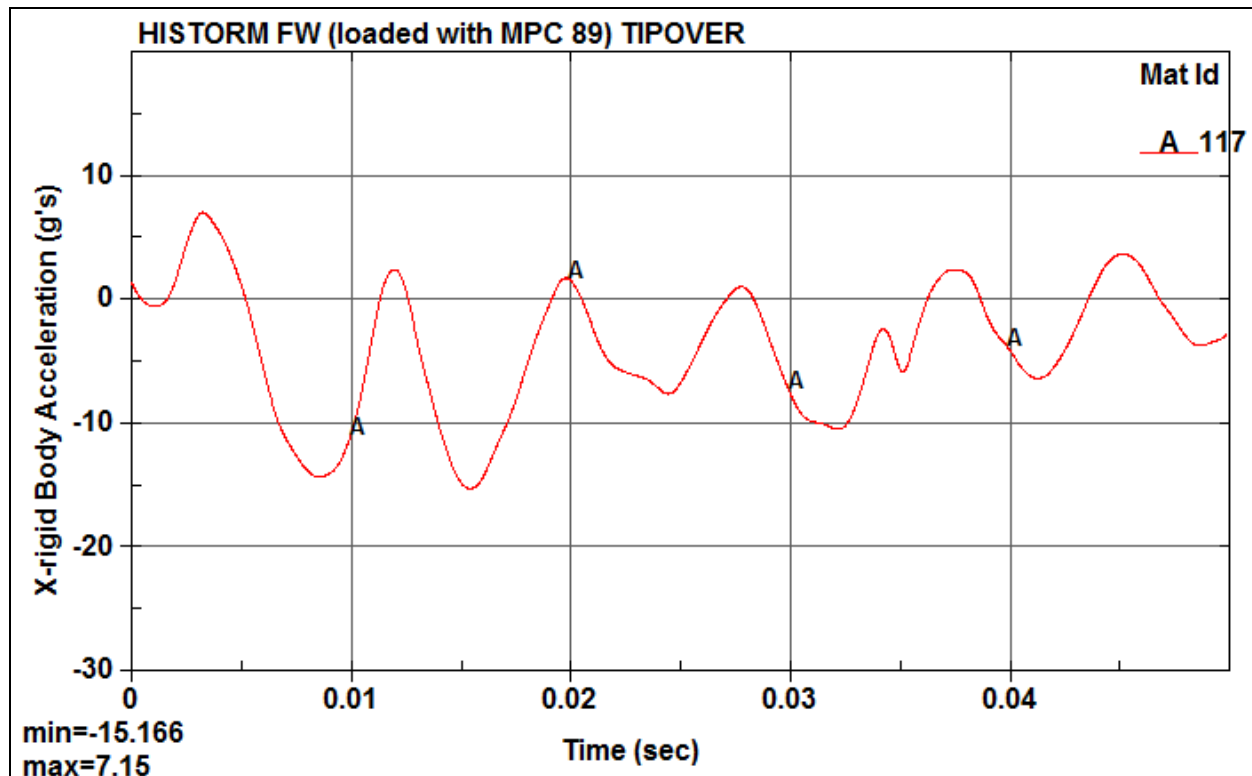
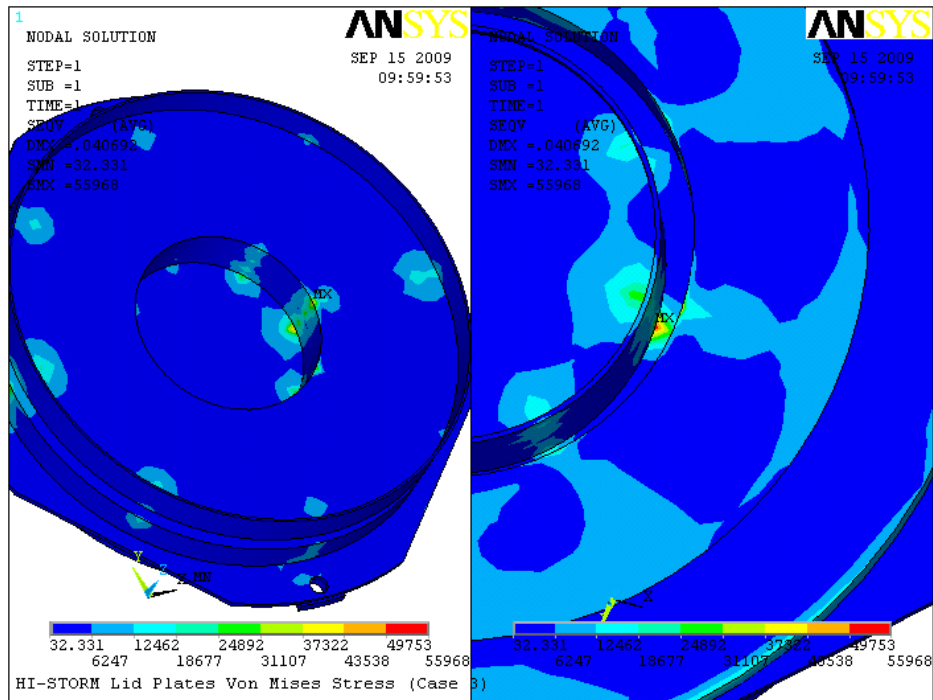
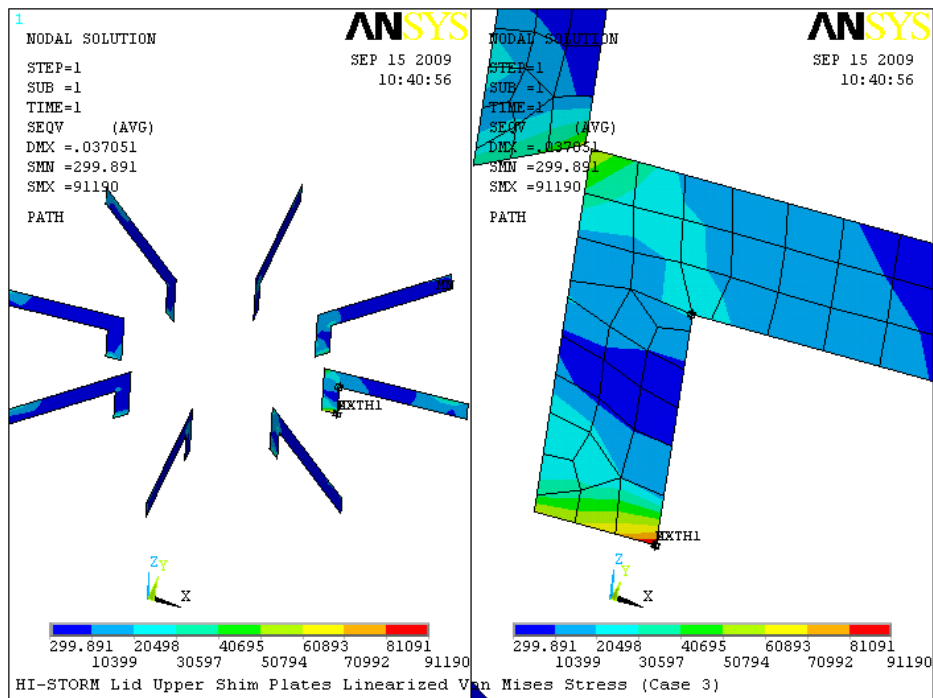


Figure 3.4.20B: Horizontal Rigid Body Deceleration Time History –  
Cask Lid Concrete (for HI-STORM FW Loaded with MPC-89)





(a) Steel Weldment (Excluding Upper Shim Plates)



(b) Upper Shim Plates

Figure 3.4.21: Stress Distribution in HI-STORM FW Lid – Non-Mechanistic Tipover

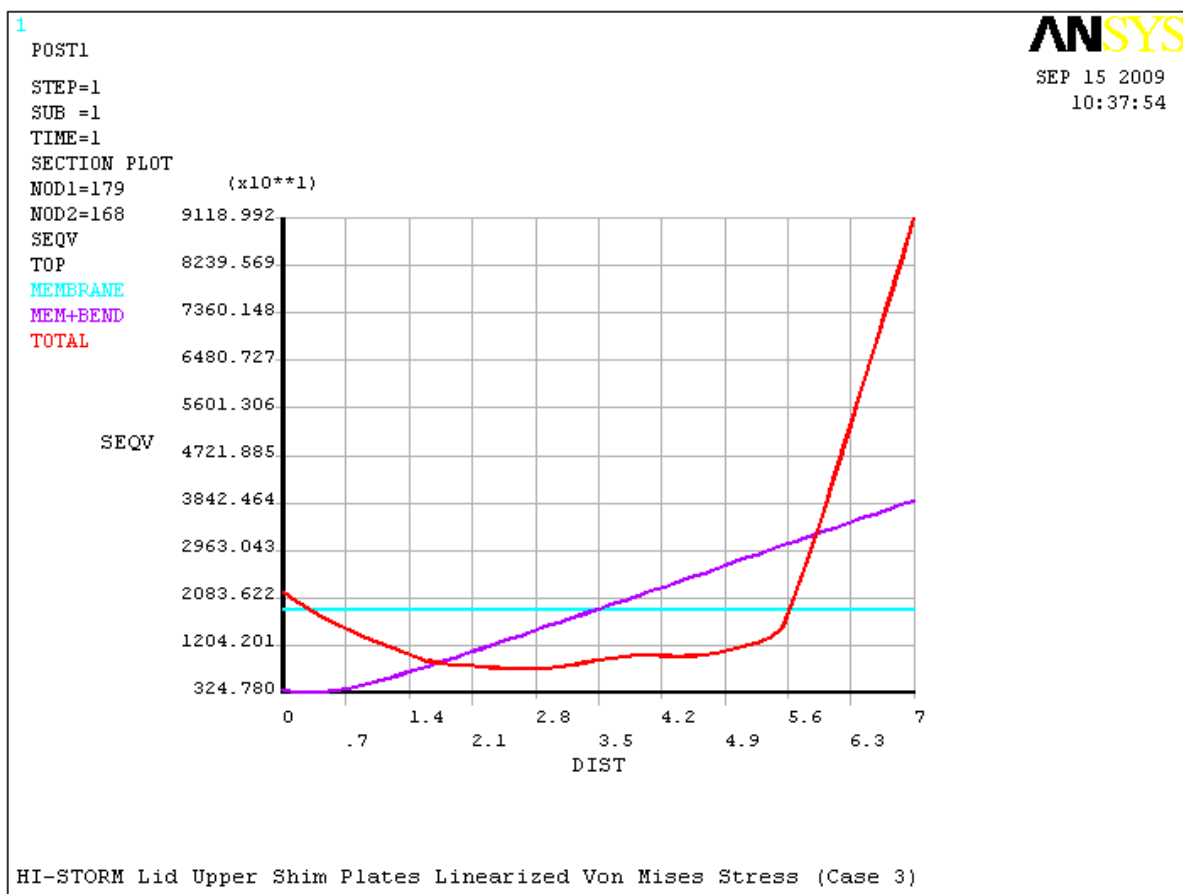


Figure 3.4.22: Linearized Stress Results for Upper Shim Plate – Non-Mechanistic Tipover

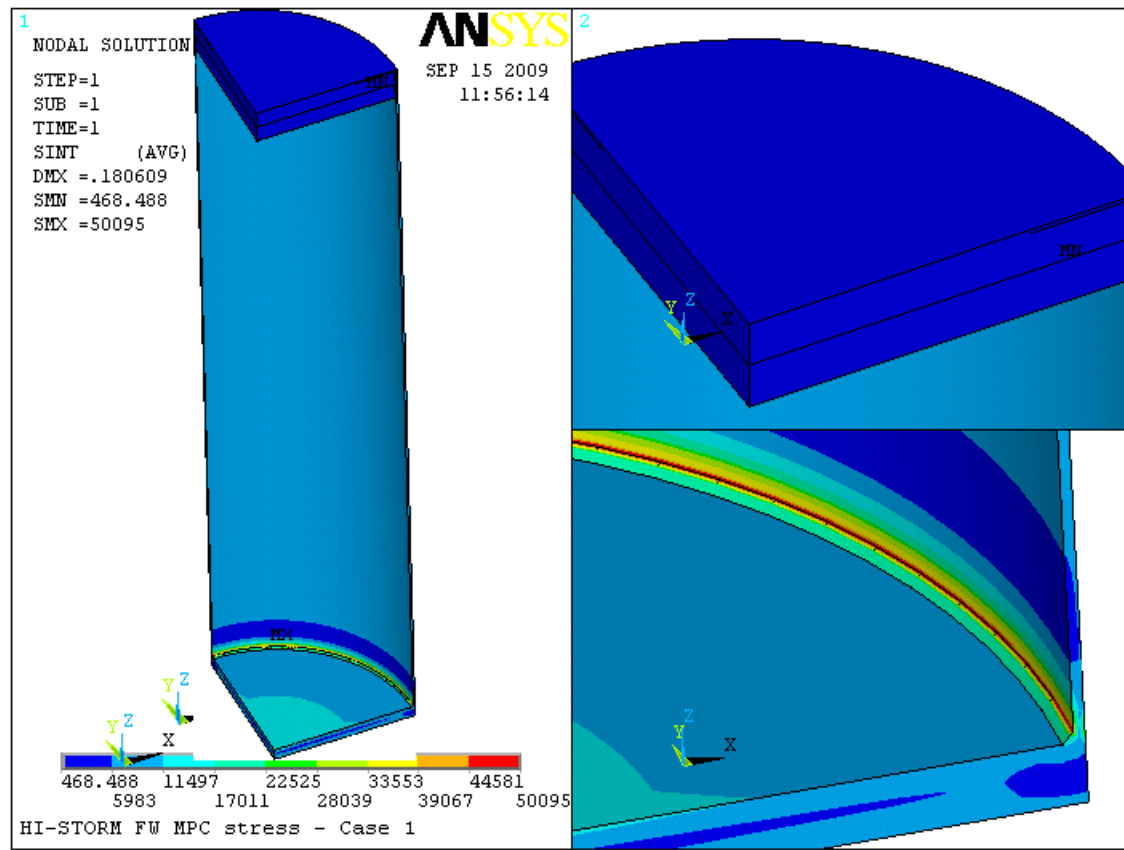


Figure 3.4.23: Stress Intensity Distribution in MPC Enclosure Vessel – Design Internal Pressure

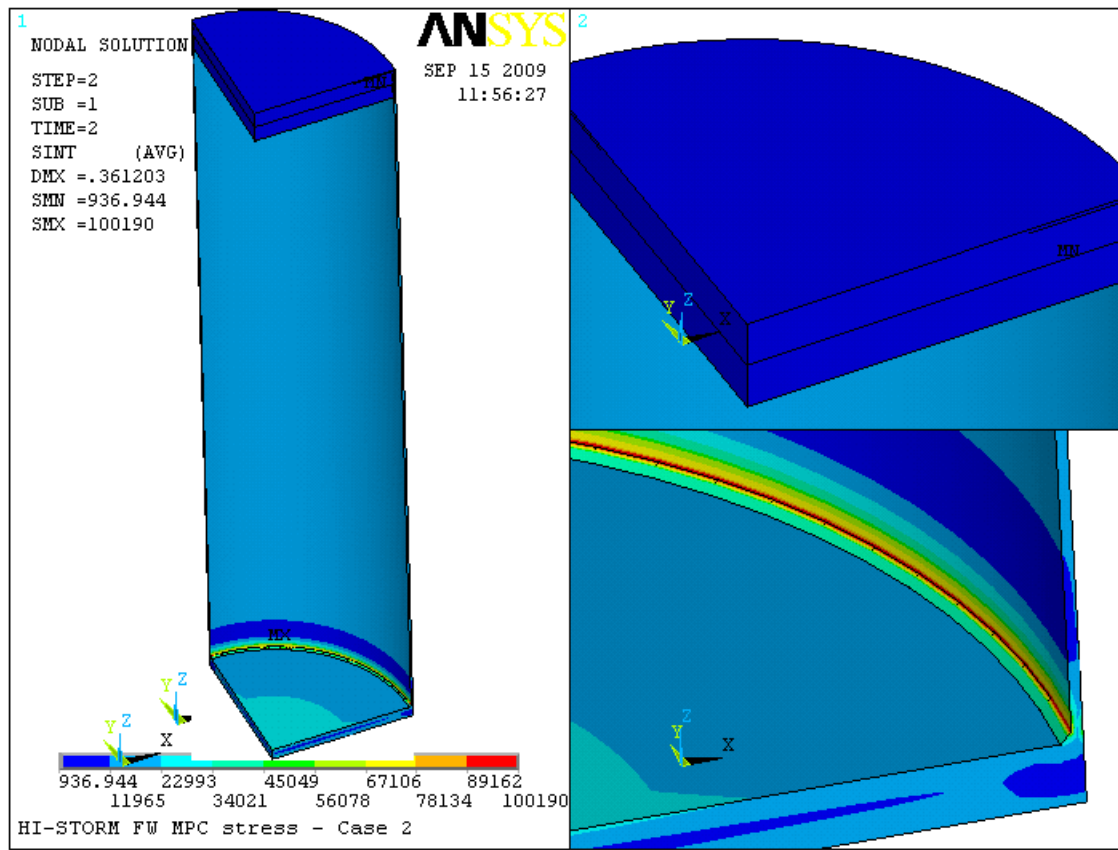
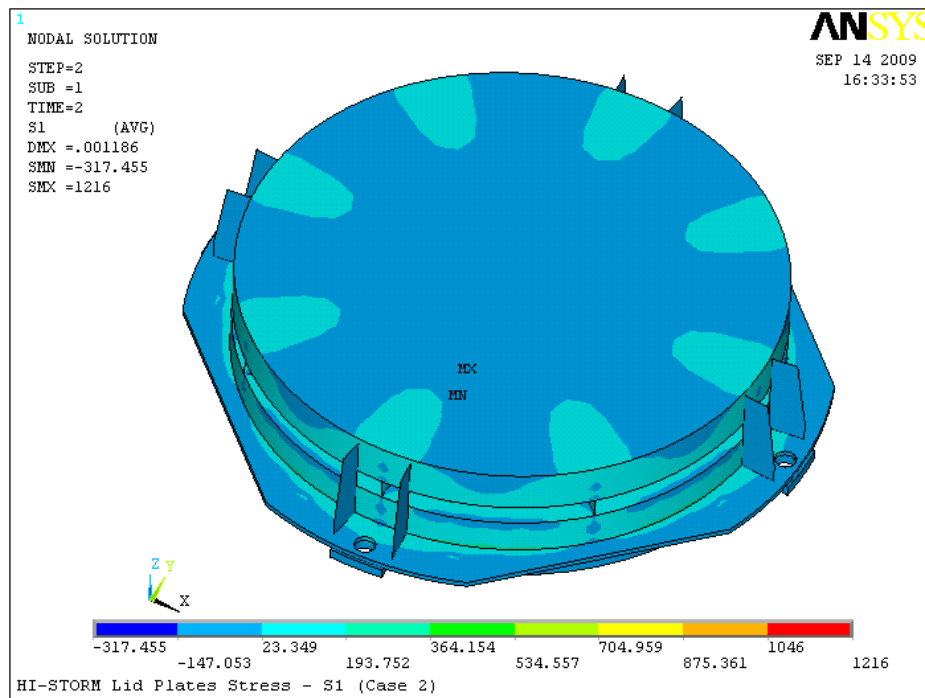
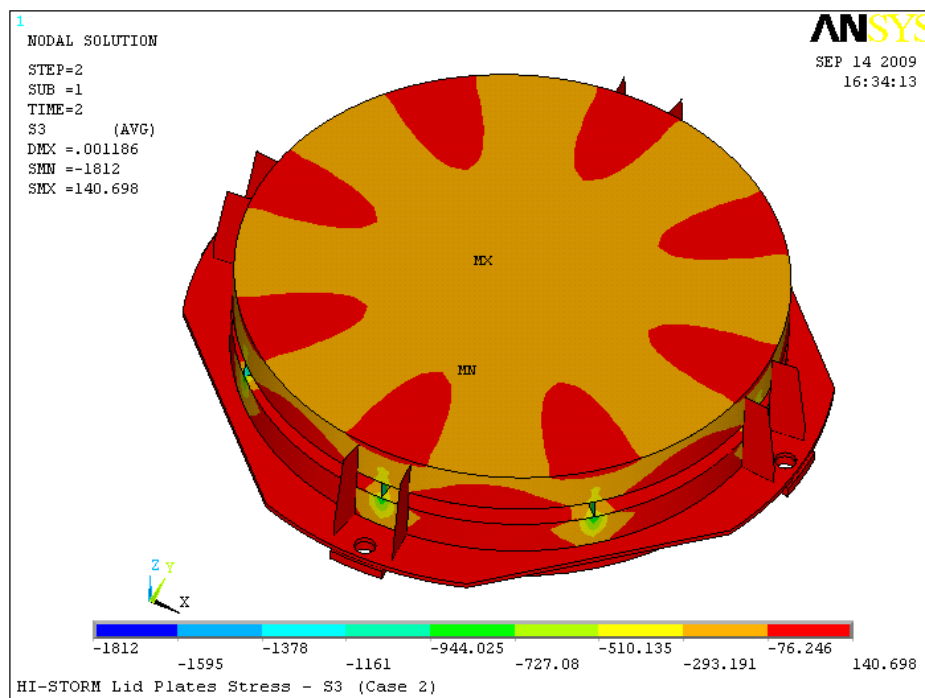


Figure 3.4.24: Stress Intensity Distribution in MPC Enclosure Vessel – Accident Internal Pressure



(a) S1 Principal Stress



(b) S3 Principal Stress

Figure 3.4.25: Stress Distribution in HI-STORM FW Lid – Snow Load

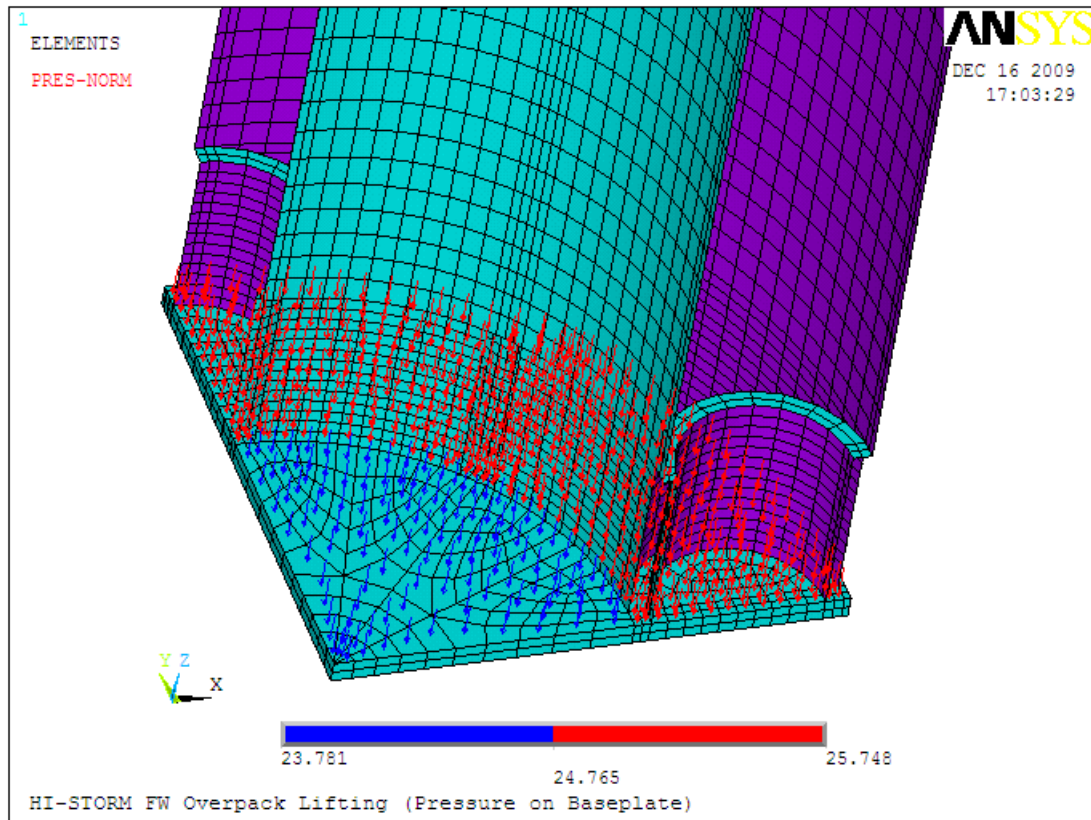


Figure 3.4.26: Applied Pressure on HI-STORM Baseplate  
Simulating Concrete Shielding and Loaded MPC

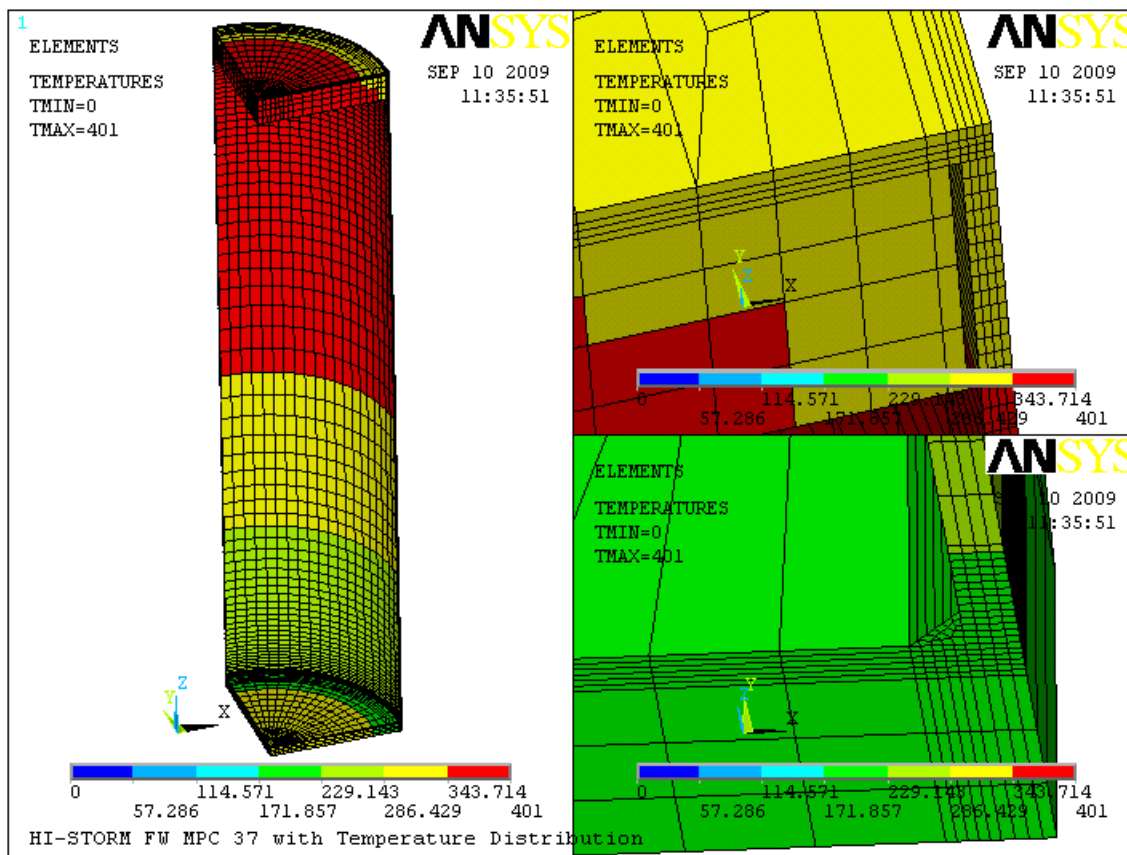


Figure 3.4.27: Normal Operating Temperature Distribution in MPC Enclosure Vessel



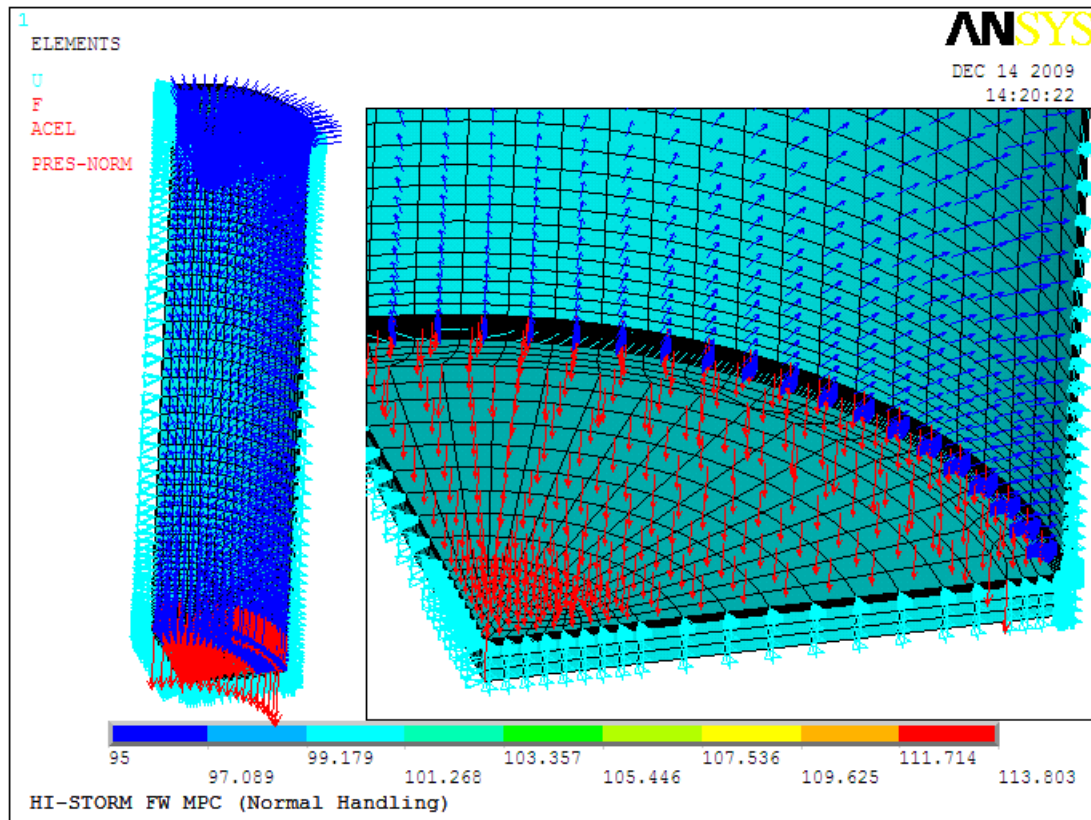


Figure 3.4.28: Normal Handling of MPC Enclosure Vessel –  
Boundary Conditions and Applied Loads



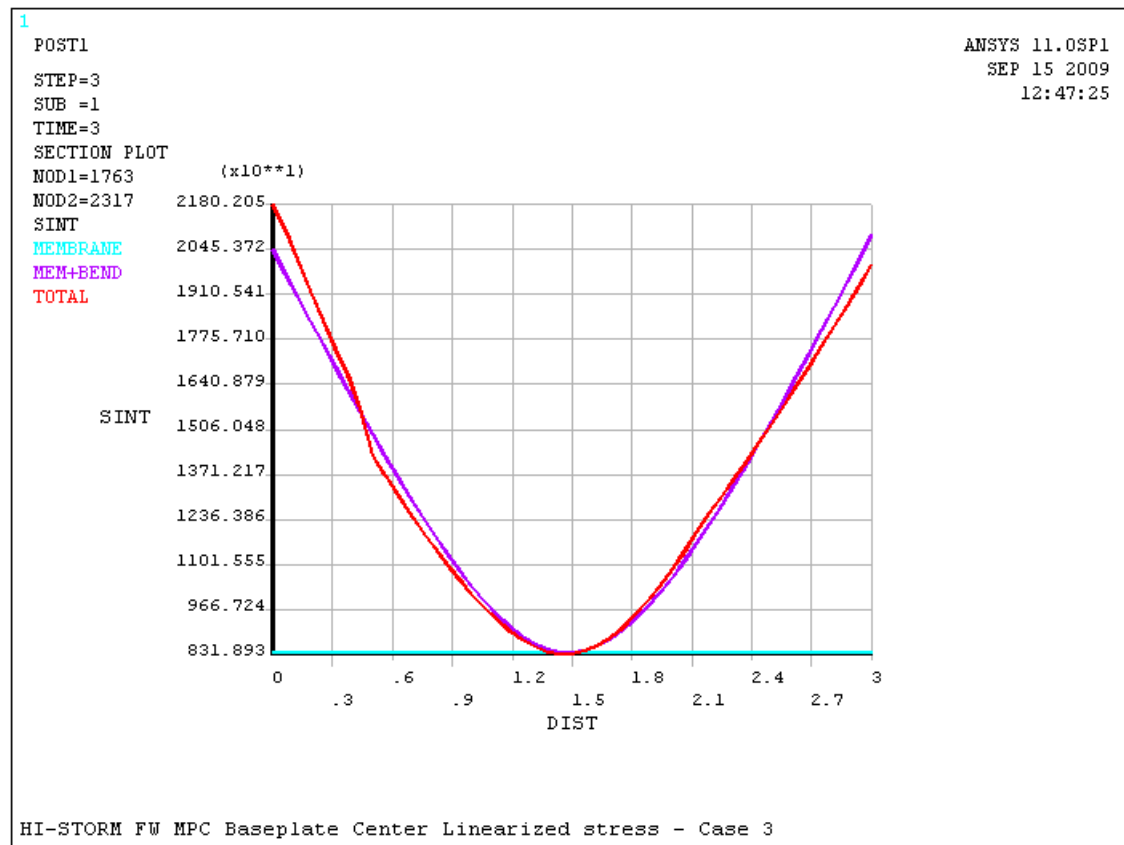


Figure 3.4.29: Normal Handling of MPC Enclosure Vessel –  
Thru-Thickness Stress Intensity Plot at Baseplate Center

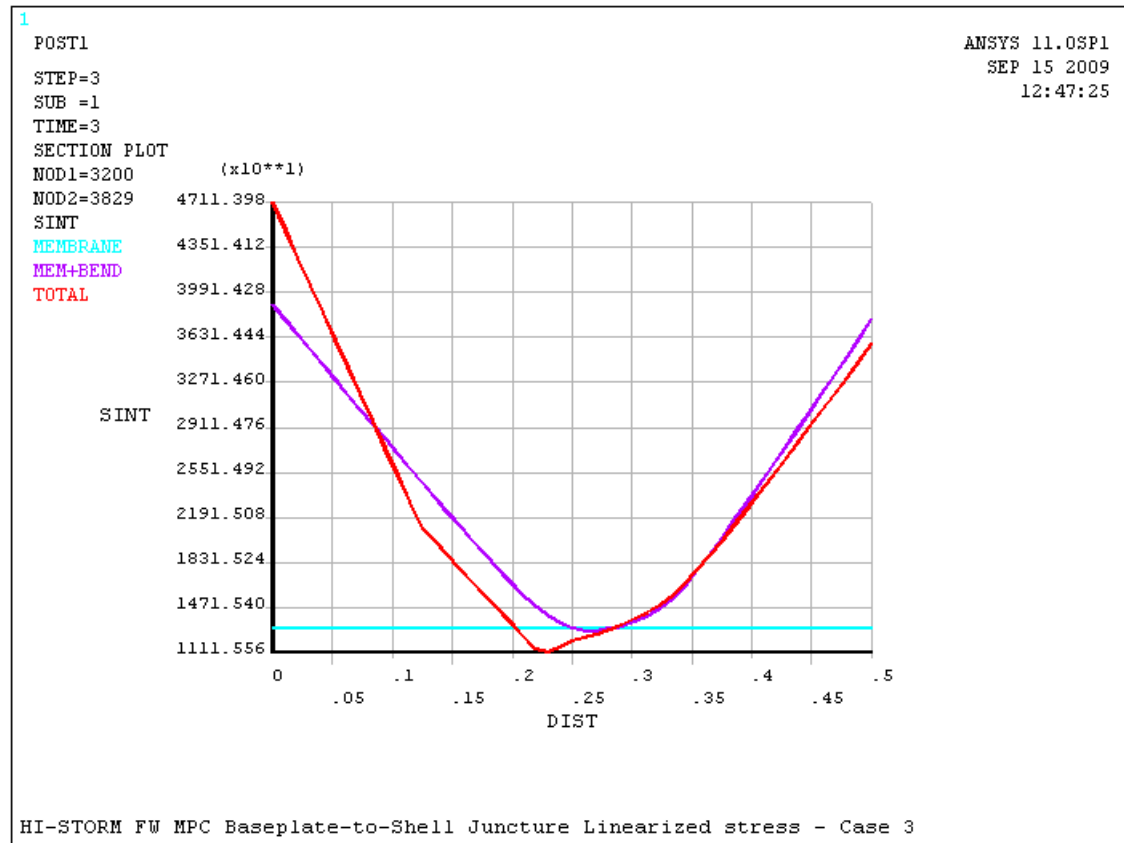


Figure 3.4.30: Normal Handling of MPC Enclosure Vessel –  
Thru-Thickness Stress Intensity Plot at Baseplate-to-Shell Juncture

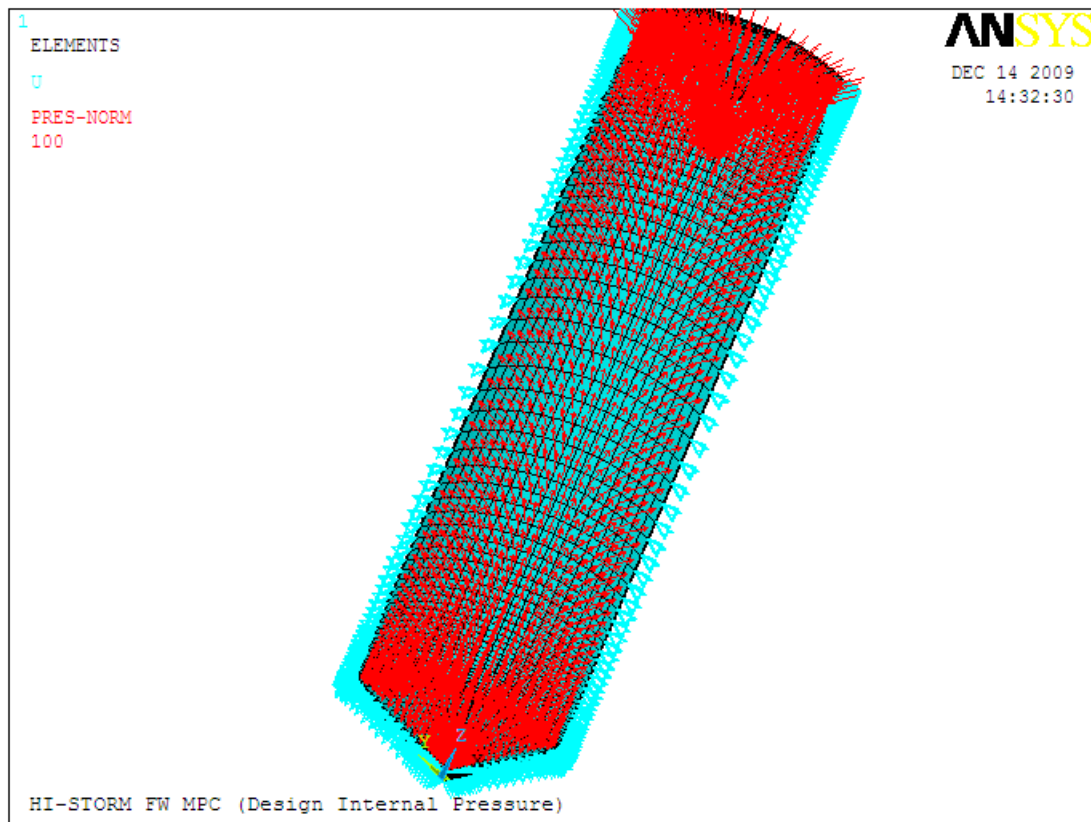


Figure 3.4.31: MPC Design Internal Pressure (Load Case 5) –  
Boundary Conditions and Applied Loads

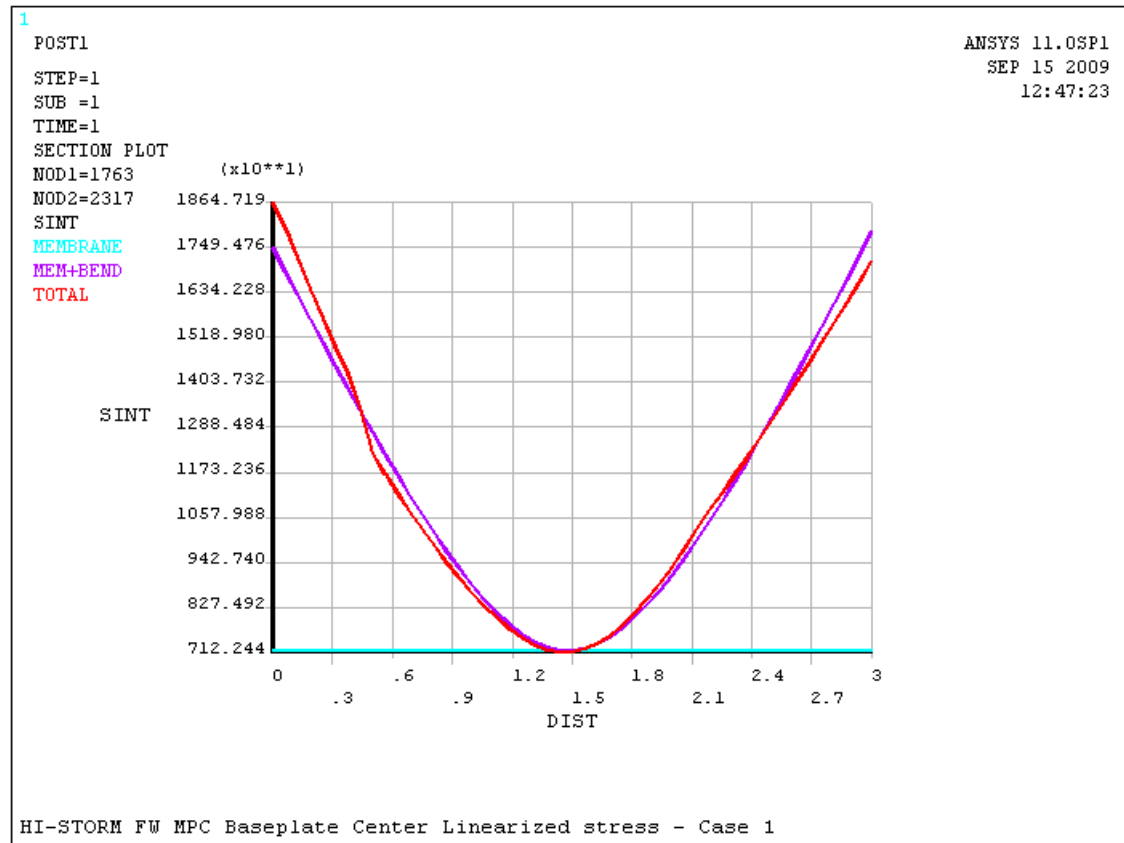


Figure 3.4.32: MPC Design Internal Pressure (Load Case 5) –  
Thru-Thickness Stress Intensity Plot at Baseplate Center

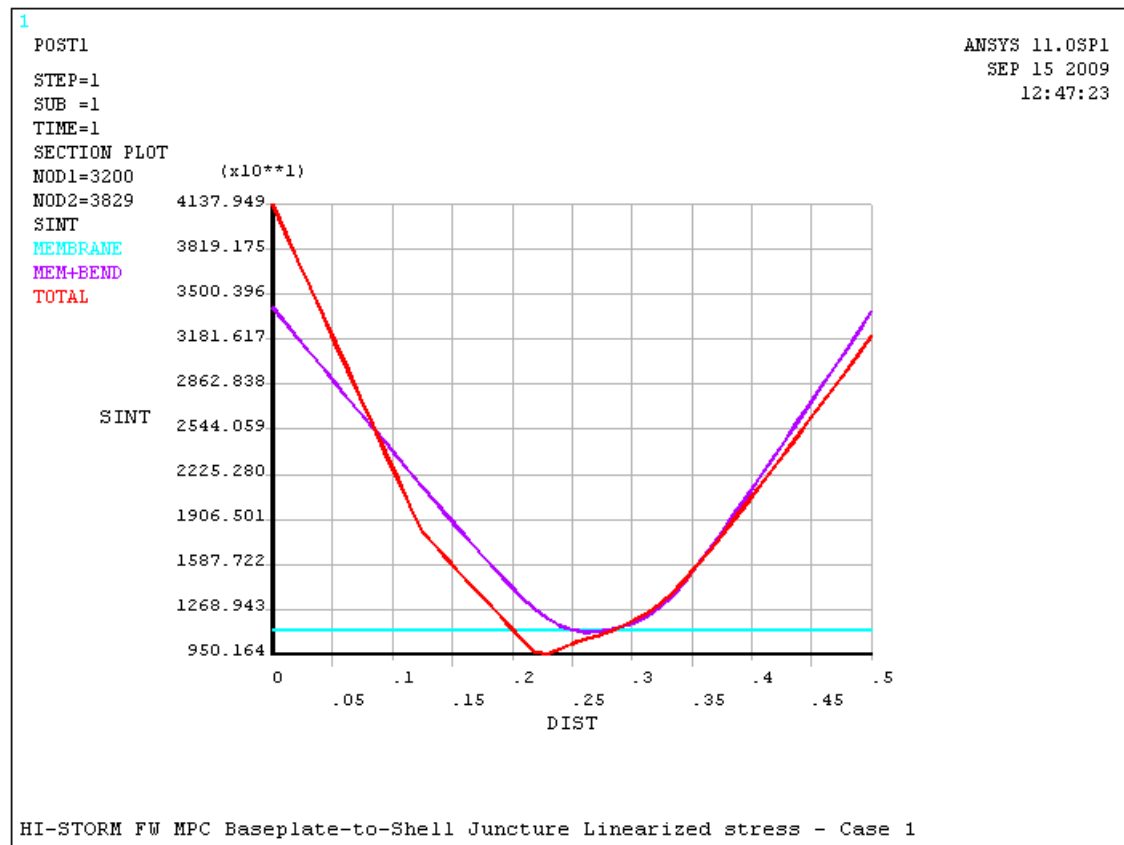


Figure 3.4.33: MPC Design Internal Pressure (Load Case 5) – Thru-Thickness Stress Intensity Plot at Baseplate-to-Shell Juncture

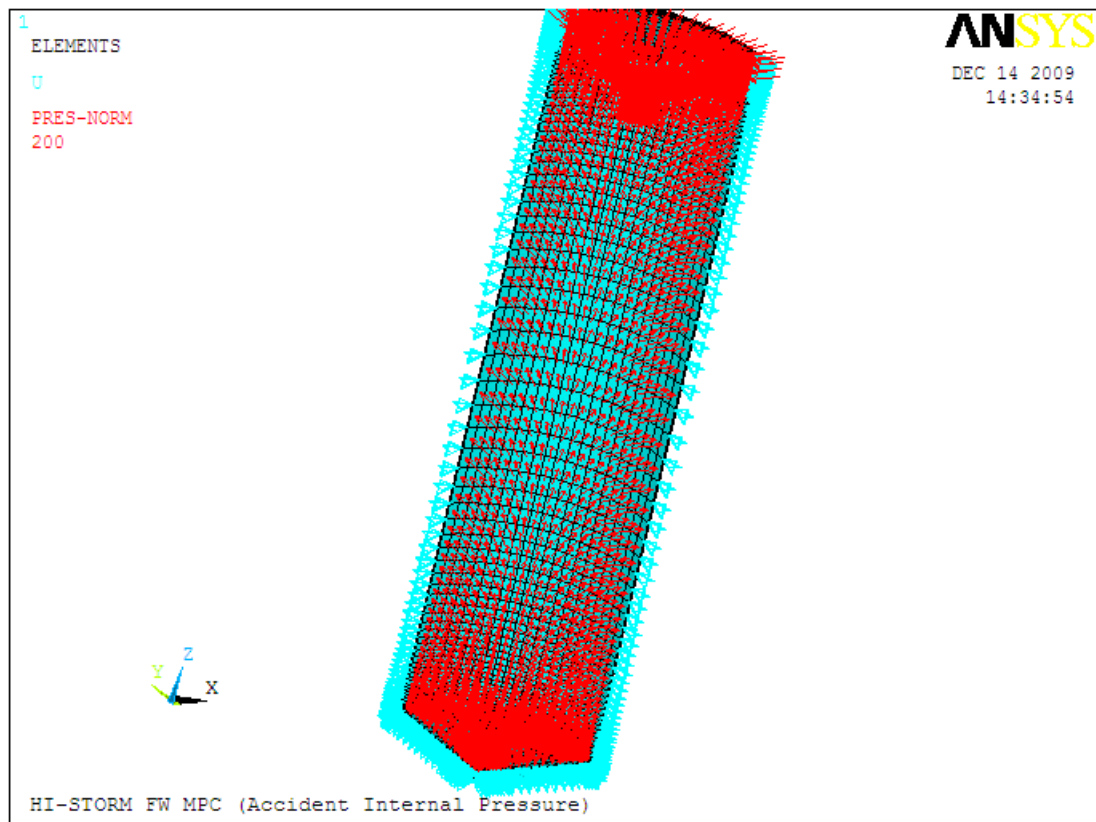


Figure 3.4.34: MPC Accident Internal Pressure (Load Case 6) –  
Boundary Conditions and Applied Loads

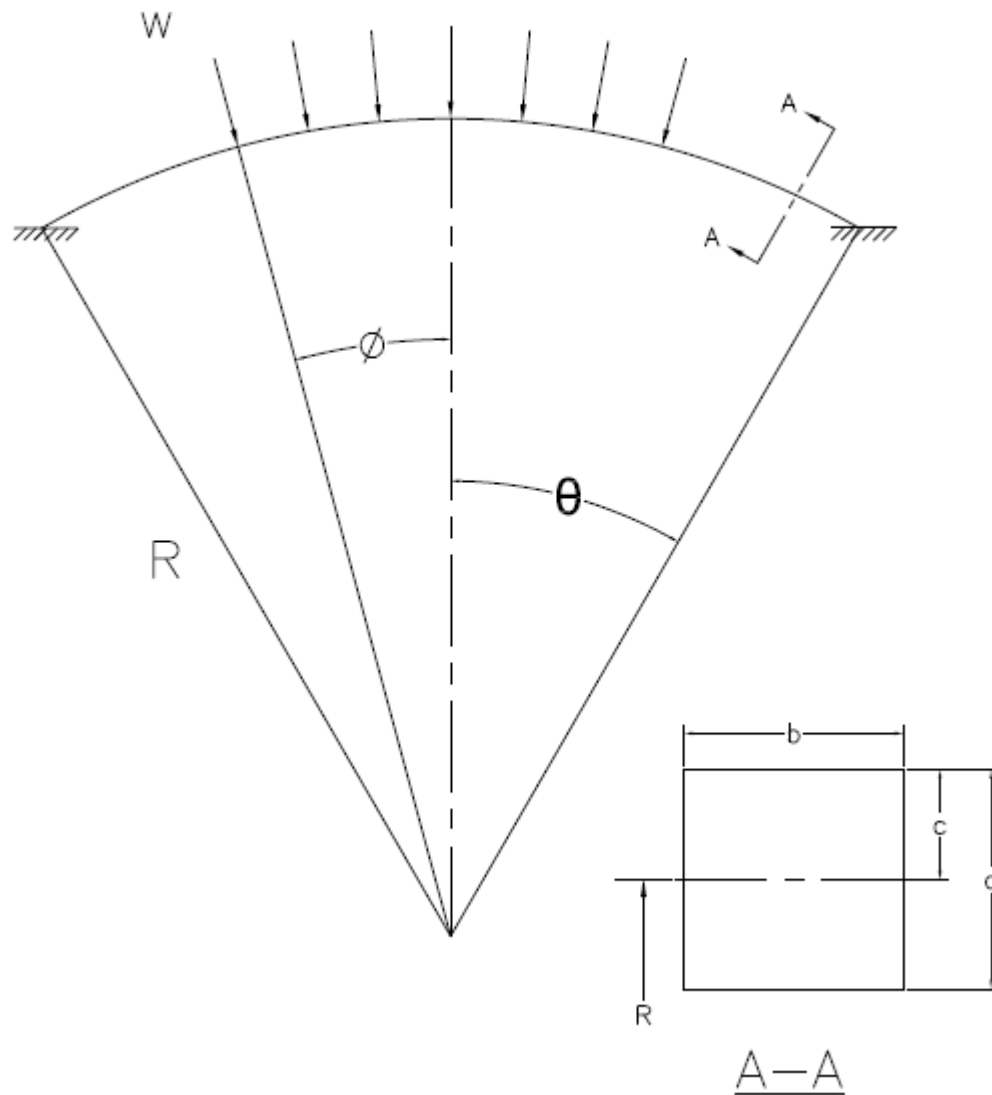


Figure 3.4.35: Analytical Model of HI-TRAC Water Jacket Shell (Load Case 8)

Figure 3.4.36

Intentionally Deleted



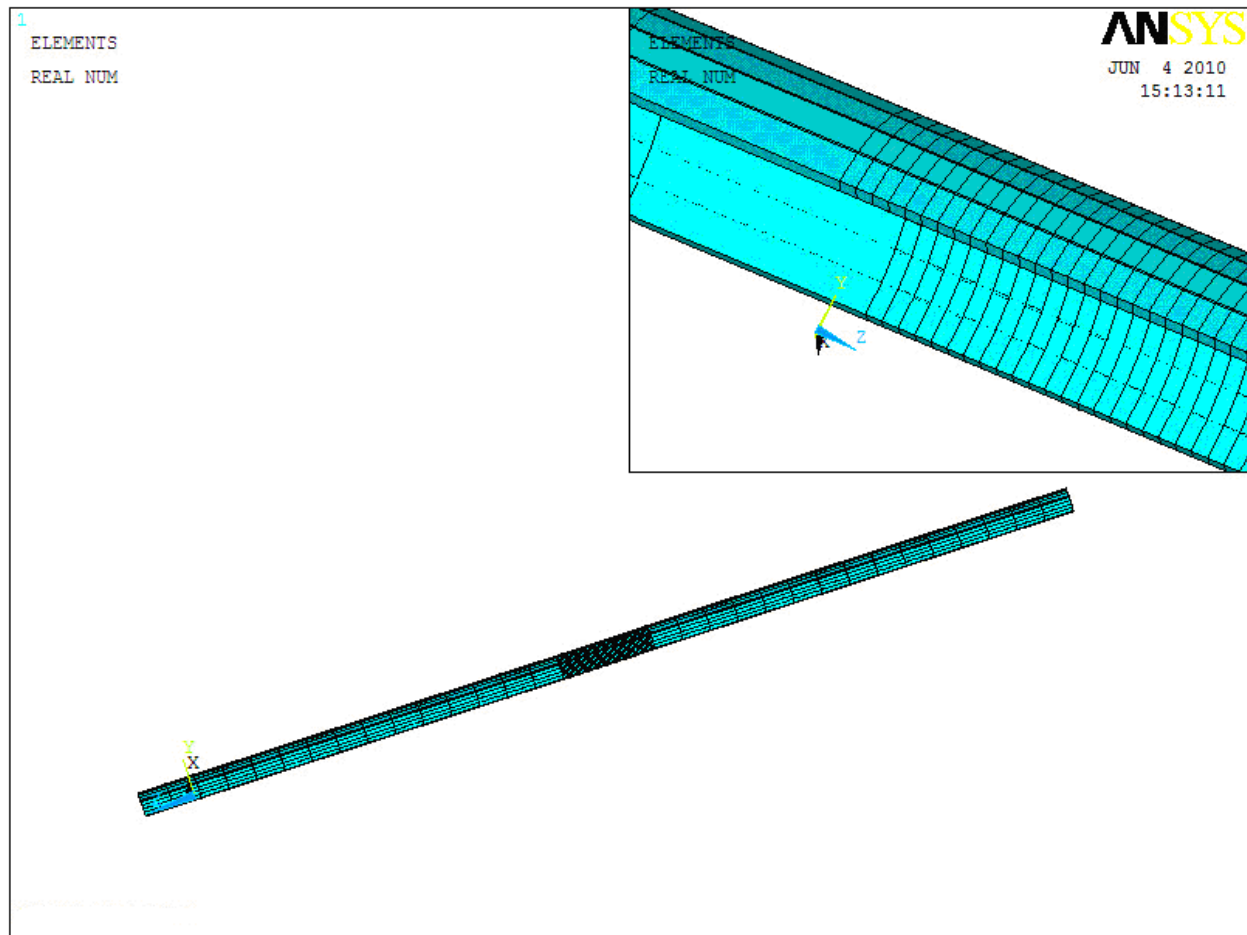


Figure 3.4.37: Finite Element Model for Fuel Rod Integrity Analysis (Load Case 11)

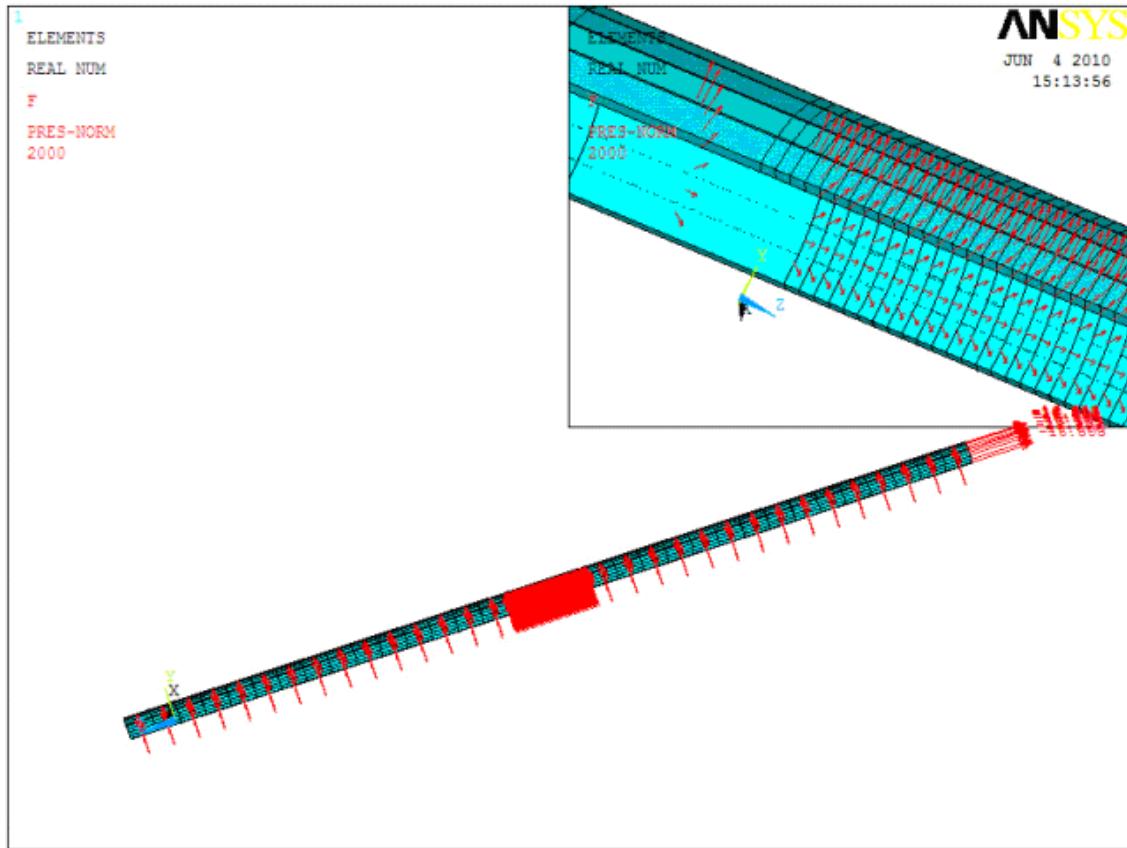


Figure 3.4.38: Applied Loads for Fuel Rod Integrity Analysis (Load Case 11)

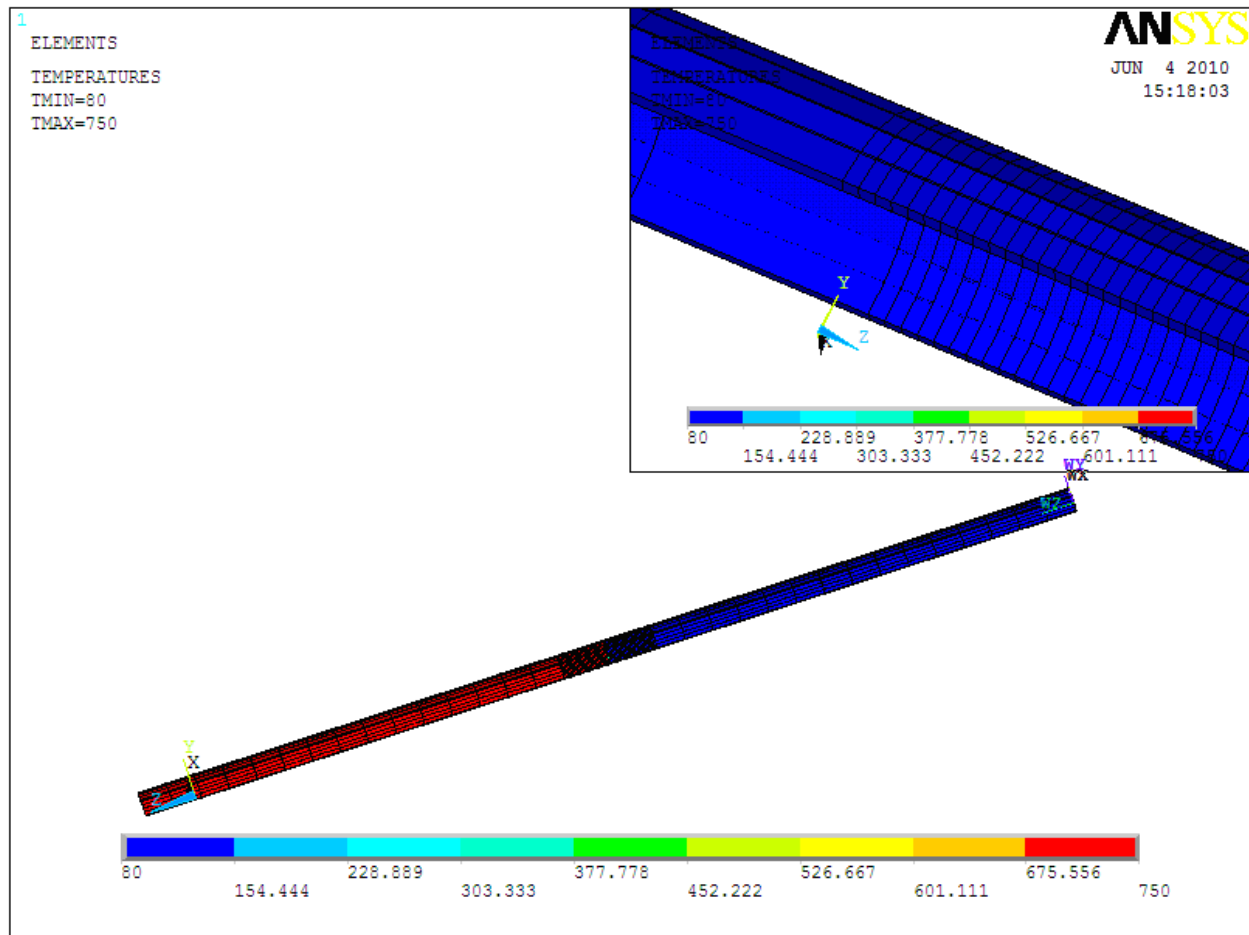


Figure 3.4.39: Applied Temperatures for Fuel Rod Integrity Analysis (Load Case 11)

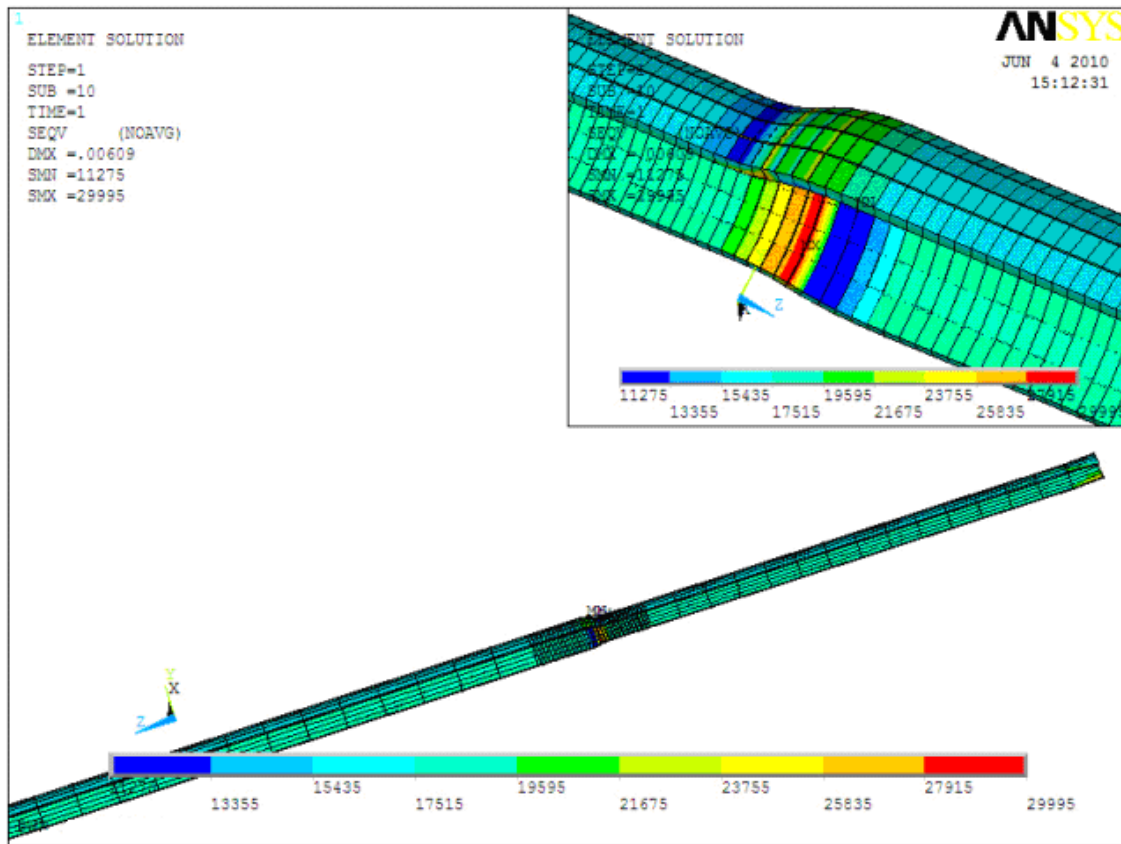


Figure 3.4.40: Stress Distribution in Fuel Rod Due to MPC Reflood (Load Case 11)

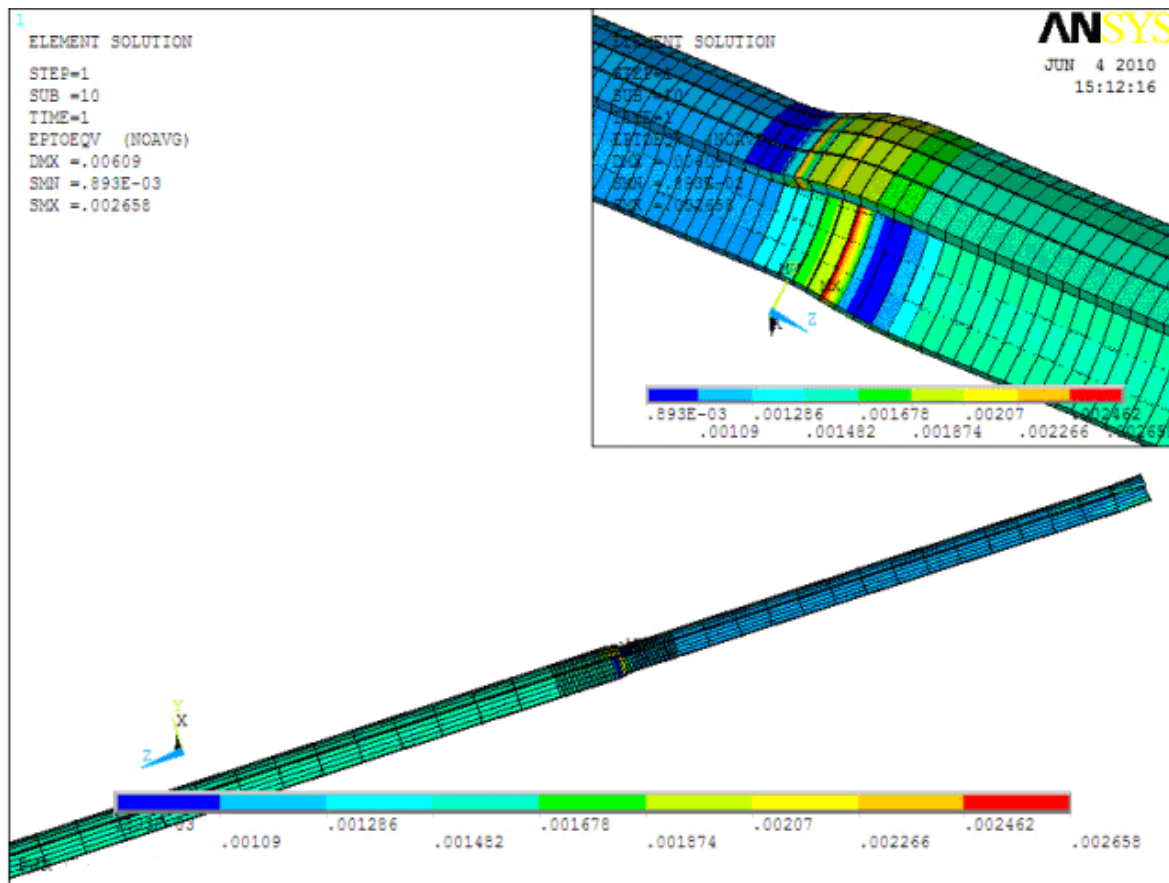


Figure 3.4.41: Strain Distribution in Fuel Rod Due to MPC Reflood (Load Case 11)

### 3.5 FUEL RODS

The regulations governing spent fuel storage cask approval and fabrication (10 CFR 72.236) require that a storage cask system “will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions” (§72.236(l)). Although the cladding of intact fuel rods does provide a barrier against the release of radioactive fission products, the confinement evaluation for the HI-STORM FW system (Chapter 7) takes no credit for fuel cladding integrity in satisfying the regulatory confinement requirement.

As described in Section 7.1, the Confinement Boundary in the HI-STORM FW system consists of the MPC Enclosure Vessel. The Enclosure Vessel is designed and, to the extent practicable, manufactured in accordance with the most stringent ASME B&PV Code (Section III, Subsection NB). As required by NB, all materials are 100% UT inspected and all butt welds are subjected to 100% volumetric inspection. The field closure features redundant barriers (the MPC lid and port cover plates are the primary barriers, the closure ring is the secondary barrier). Section 7.1 further describes that the MPC design, welding, testing and inspection requirements meet the guidance of ISG-18 [7.1.2] such that leakage from the Confinement Boundary is non-credible. Section 7.2 addresses confinement for normal and off-normal conditions, and concludes that since the MPC confinement vessel remains intact, and the design bases temperatures and pressure are not exceeded, leakage from the MPC Confinement Boundary is not credible. Confinement for accident conditions is addressed in Section 7.3, which concludes that there is no mechanistic failure mode that could result in a breach of the Confinement Boundary, and escape of radioactive materials to the environment.

Since fuel rod cladding is not considered in the design criteria for the confinement of radioactive material under normal, off-normal, or accident conditions of storage, no specific analysis or test results are required to demonstrate cladding integrity.

## 3.6 SUPPLEMENTAL DATA

### 3.6.1 Calculation Packages

In addition to the calculations presented in Chapter 3, supporting calculation packages have been prepared to document other information pertinent to the analyses. Supporting calculation packages back up the summary results reported in the FSAR. The Calculation Packages are referenced in the body of the FSAR and are maintained as proprietary documents in Holtec's Configuration Control system.

### 3.6.2 Computer Programs

Two 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 FSAR. These codes are ANSYS and LS-DYNA. A third computer program, Visual Nastran, is also described below even though it is not explicitly used in this FSAR. It may, however, be used to perform the seismic stability evaluation of HI-STORM FW casks for a specific ISFSI site where NUREG/CR-6865 is not applicable (see Subsection 3.4.4.1.2).

#### i. ANSYS Mechanical

ANSYS is the original (and commonly used) name for ANSYS Mechanical general-purpose finite element analysis software. ANSYS Mechanical is the version of ANSYS commonly used for structural applications. It is a self contained analysis tool incorporating pre-processing (geometry creation, meshing), solver, and post processing modules in a unified graphical user interface. ANSYS Mechanical is a general purpose finite element modeling package for numerically solving a wide variety of mechanical problems. These problems include: static/dynamic structural analysis (both linear and non-linear), heat transfer and fluid problems, as well as acoustic and electro-magnetic problems.

ANSYS Mechanical has been independently QA validated by Holtec International and used for structural analysis of casks, fuel racks, pressure vessels, and a wide variety of SSCs, for over twenty years.

#### ii. LS-DYNA

LS-DYNA is a general purpose finite element code for analyzing the large deformation static and dynamic response of structures including structures coupled to fluids. The main solution methodology is based on explicit time integration and is therefore well suited for the examination of the response to shock loading. A contact-impact algorithm allows difficult contact problems to be easily treated. Spatial discretization is achieved by the use of four node tetrahedron and eight node solid elements, two node beam elements, three and four node shell elements, eight node solid shell

elements, truss elements, membrane elements, discrete elements, and rigid bodies. A variety of element formulations are available for each element type. Adaptive re-meshing is available for shell elements. LS-DYNA currently contains approximately one hundred constitutive models and ten equations-of-state to cover a wide range of material behavior.

In this safety analysis report, LS-DYNA is used to analyze all loading conditions that involve short-time dynamic effects.

LS-DYNA is maintained in a QA-validated status in Holtec's Configuration Control system.

iii. Visual Nastran

Visual Nastran [3.6.1] is used for rigid body motion simulation of the cask components, where a simplified analysis is appropriate. VisualNastran is a kinematics simulation code that includes large orientation change capability, simulation of impacts, and representation of contact and friction behavior. Visual Nastran Desktop (VN) performs time history dynamic analysis of freestanding structures using the acceleration time-histories in the three orthogonal directions as the input. It provides a complete articulation of the dynamic response of the rigid body, including sliding, precession, and tipping (and combinations thereof). Visual Nastran is maintained in a QA-validated status by Holtec International.

All three computer codes have been benchmarked and QA-validated to establish their veracity.

The compliance matrix below provides the necessary information to document their validation status, and the measures employed pursuant to ISG-21 and Holtec's QA program, to ensure error-free solutions.



<b>ISG-21 and QA Compliance Matrix for Computer Codes</b>				
	Item	ANSYS	LS-DYNA	Visual Nastran
1.	Benchmark and QA-validation are documented in Holtec Report No.(s) (Proprietary Reports)	HI-2012627	HI-961519	HI-2022896
2.	Computer Program Type (Public or Private Domain)	Public Domain	Public Domain	Public Domain
3.	Does Holtec maintain a system evaluating error notices if any are issued by the Code provider to evaluate their effect on the safety analyses carried out using the Code, including Part 21 notification? (Yes/No)	Yes	Yes	Yes
4.	Is the use of the Code restricted to personnel qualified under the Company's personnel qualification program? (Yes/No)	Yes	Yes	Yes
5.	Has benchmarking been performed against sample problems with known independently obtained numerical solutions (Yes/No)	Yes	Yes	Yes
6.	Have element types used in the safety analyses herein also employed in the benchmarking effort? (Yes/No)	Yes	Yes	N/A
7.	Are the element types used in this FSAR also used in other Holtec dockets that support other CoCs? (Yes/No)	Yes	Yes	N/A
8.	Is each update of the Code vetted for backwards consistency with prior updates? (Yes/No)	Yes	Yes	Yes
9.	Is the use of the Code limited to the range of parameters specified in the User Manual provided by the Code Developer? (Yes/No)	Yes	Yes	Yes
10.	Are the element aspect ratios, where applicable, used in the simulation model within the limit recommended by the Code Developer or Holtec's successful experience in other safety analyses? (Yes/No)	Yes	Yes	N/A
11.	Are element sizes used in the simulation models consistent with past successful analyses in safety significant applications? (Yes/No)	Yes	Yes	N/A
12.	Was every computer run in this chapter free of an error warning (i.e., in hidden warnings in the Code that indicate a possible error in the solution)? (Yes/No)	Yes	Yes	N/A
13.	If the answer to the above is No, then is the annotated warning discussed in the discussion of the result in this report?	N/A	N/A	N/A

### 3.7 COMPLIANCE WITH THE STRUCTURAL REQUIREMENTS IN PART 72

Supporting information to provide reasonable assurance with respect to the adequacy of the HI-STORM FW system to store spent nuclear fuel in accordance with the stipulations of 10CFR72 is presented throughout this FSAR. The following statements are applicable to an affirmative structural safety evaluation:

- The design and structural analysis of the HI-STORM FW system is in compliance with the provisions of Chapter 3 of NUREG-1536 as applicable.
- The HI-STORM FW structures, systems, and components (SSC) that are important to safety (ITS) are identified in the licensing drawings in Section 1.5. The licensing drawings present the HI-STORM FW SSCs in adequate detail and the explanatory narratives in Sections 3.1 and 3.4 provide sufficient textual details to allow an independent evaluation of their structural effectiveness.
- The requirements of 10CFR72.24 with regard to information pertinent to structural evaluation is provided in Chapters 2, 3, and 12.
- Technical Specifications pertaining to the structures of the HI-STORM FW system have been provided in Chapter 13 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 in the HI-STORM FW system 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-STORM FW/HI-TRAC VW overpacks for normal, off-normal, accident, and natural phenomenon events are provided in Subsection 3.1.2.2.
  - ii. Stress limits applicable to the Code materials are found in Section 3.3.
  - iii. The stress and displacement response of the fuel basket, the enclosure vessel, and the HI-STORM FW/HI-TRAC VW overpacks for all applicable loads have been computed by analysis and reported in Subsections 3.4.3 and 3.4.4. Descriptions of stress analysis models are presented in Subsection 3.1.3.
- The structural design and fabrication details of the fuel baskets whose safety function

in the HI-STORM FW system is to maintain nuclear criticality safety, are provided in the drawings in Section 1.5. The structural factors of safety, summarized in Section 3.4 for all credible load combinations under normal, off-normal, accident, and natural phenomenon events demonstrate that the acceptance criteria are satisfied in all cases. In particular, the maximum lateral deflection in the fuel basket panels under accident events has been determined to be within the limit used in the criticality analysis (see Subsection 3.4.4.1.4). Thus, the requirement of 10CFR72.124(a), with respect to structural margins of safety for SSCs important to nuclear criticality safety are fully satisfied.

- Structural margins of safety during handling, packaging, and transfer operations, under the provisions of 10CFR Part 72.236(b), imply that the lifting and handling devices be engineered to comply with the stipulations of ANSI N14.6 and NUREG-0612, as applicable. The requirements of the governing standards for handling operations are summarized in Subsection 3.4.3 herein. Factors of safety for all ITS components under lifting and handling operations are summarized in tables in Section 3.4, which show that adequate structural margins exist in all cases.
- Consistent with the provisions of 10CFR72.236(i), the Confinement Boundary for the HI-STORM FW 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 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 primary lid of the MPC Enclosure Vessel is welded using a strength groove weld and is subjected to multiple liquid penetrant examinations and pressure testing to establish a maximum confidence in weld joint integrity.
  - iii. The closure system 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 the environmental conditions of an ISFSI.
  - v. The load combinations for normal, off-normal, accident, and natural phenomena events have been compiled and applied on the MPC Enclosure Vessel (Confinement Boundary). The results, summarized in Section 3.4, show that the factor of safety (with respect to the appropriate limits) is

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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 reaching and breaching the Confinement Boundary. Analyses presented in Section 3.4 and supplemented by Appendices 3.A and 3.B show that the integrity of 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.
- The structural design features in the HI-STORM FW system are in compliance with the specific requirements of 10CFR72.236(e), (f), (g), (h), (i), (j), (k), and (m).

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APPENDIX 3.A – RESPONSE OF HI-STORM FW AND HI-TRAC VW  
TO TORNADO WIND LOAD AND LARGE MISSILE IMPACT

**Withheld in Accordance with 10 CFR 2.390**

APPENDIX 3.B – MISSILE PENETRATION ANALYSES  
FOR HI-STORM FW AND HI-TRAC VW

**Withheld in Accordance with 10 CFR 2.390**

APPENDIX 3.C – CODE CASE N-284-2 STABILITY CALCULATIONS FOR MPC SHELL

**Withheld in Accordance with 10 CFR 2.390**



## CHAPTER 4\* THERMAL EVALUATION

### 4.0 OVERVIEW

The HI-STORM FW system is designed for long-term storage of spent nuclear fuel (SNF) in a vertical orientation. The design envisages an array of HI-STORM FW systems laid out in a rectilinear pattern stored on a concrete ISFSI pad in an open environment. In this chapter, compliance of HI-STORM FW system's thermal performance to 10CFR72 requirements for outdoor storage at an ISFSI using 3-D thermal simulation models is established. The analyses consider passive rejection of decay heat from the stored SNF assemblies to the environment under normal, off-normal, and accident conditions of storage. Finally, the thermal margins of safety for long-term storage of both moderate burnup (up to 45,000 MWD/MTU) and high burnup spent nuclear fuel (greater than 45,000 MWD/MTU) in the HI-STORM FW system are quantified. Safe thermal performance during on-site loading, unloading and transfer operations, collectively referred to as "short-term operations" utilizing the HI-TRAC VW transfer cask is also evaluated.

The HI-STORM FW thermal evaluation follows the guidelines of NUREG-1536 [4.4.1] and ISG-11 [4.1.4]. These guidelines provide specific limits on the permissible maximum cladding temperature in the stored commercial spent fuel (CSF)<sup>†</sup> and other Confinement Boundary components, and on the maximum permissible pressure in the confinement space under certain operating scenarios. Specifically, the requirements are:

1. The fuel cladding temperature must meet the temperature limit under normal, off-normal and accident conditions appropriate to its burnup level and condition of storage or handling set forth in Table 4.3.1.
2. The maximum internal pressure of the MPC should remain within its design pressures for normal, off-normal, and accident conditions set forth in Table 2.2.1.
3. The temperatures of the cask materials shall remain below their allowable limits set forth in Table 2.2.3 under all scenarios.

As demonstrated in this chapter, the HI-STORM FW system is designed to comply with all of the criteria listed above. Sections 4.1 through 4.3 describe thermal analyses and input data that are common to all conditions of storage, handling and on-site transfer operations. All thermal analyses to evaluate normal conditions of storage in a HI-STORM FW storage module are described in Section 4.4. All thermal analyses to evaluate normal handling and on-site transfer in a HI-TRAC VW transfer cask are described in Section 4.5. All thermal analyses to evaluate off-normal and accident conditions are described in Section 4.6. This SAR chapter is in full

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\* 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. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. All terms-of-art used in this chapter are consistent with the terminology of the Glossary. Finally, all evaluations and results presented in this Chapter are supported by calculation packages cited herein (References [4.1.9] and [4.1.10]).

<sup>†</sup> Defined as nuclear fuel that is used to produce energy in a commercial nuclear reactor (See Glossary).

compliance with ISG-11 and with NUREG-1536 guidelines, subject to the exceptions and clarifications discussed in Chapter 1, Table 1.0.3.

As explained in Section 1.2, the storage of SNF in the fuel baskets in the HI-STORM FW system is configured for a three-region storage system. Figures 1.2.1 and 1.2.2 provide the information on the location of the regions and Tables 1.2.3 and 1.2.4 provide the permissible specific heat load (heat load per fuel assembly) in each region for the PWR and BWR MPCs, respectively. The Specific Heat Load (SHL) values are defined for two patterns that in one case maximizes ALARA (Table 1.2.3, Pattern A and Table 1.2.4) and in the other case maximizes heat dissipation (Table 1.2.3, Pattern B). The ALARA maximized fuel loading is guided by the following considerations:

- Region 1: Located in the core region of the basket is permitted to store fuel with medium specific heat load.
- Region 2: This is the intermediate region flanked by the core region (Region I) from the inside and the peripheral region (Region III) on the outside. This region has the maximum SHL in the basket.
- Region 3: Located in the peripheral region of the basket, this region has the smallest SHL. Because a low SHL means a low radiation dose emitted by the fuel, the low heat emitting fuel around the periphery of the basket serves to block the radiation from the Region II fuel, thus reducing the total quantity of radiation emanating from the MPC in the lateral direction.

Thus, the 3-region arrangement defined above serves to minimize radiation dose from the MPC and peak cladding temperatures mitigated by avoiding placement of hot fuel in the basket core.

To address the needs of cask users having high heat load fuel inventories, fuel loading Pattern B is defined to maximize heat dissipation by locating hotter fuel in the cold peripheral Region 3 and in this manner minimize cladding temperatures. This has the salutary effect of minimizing core temperature gradients in the radial direction and thermal stresses in the fuel and fuel basket.

The salutary consequences of all regionalized loading arrangements become evident from the computed peak cladding temperatures in this chapter, which show margin to the ISG-11 limit discussed earlier.

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 unquantified 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

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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.

## 4.1 DISCUSSION

The aboveground HI-STORM FW system consists of a sealed MPC situated inside a vertically-oriented, ventilated storage overpack. Air inlet and outlet ducts that allow for air cooling of the stored MPC are located at the bottom and top, respectively, of the cylindrical overpack (see Figure 4.1.1). The SNF assemblies reside inside the MPC, which is sealed with a welded lid to form the Confinement Boundary. The MPC contains a Metamic-HT egg-crate fuel basket structure with square-shaped compartments of appropriate dimensions to allow insertion of the fuel assemblies prior to welding of the MPC lid and closure ring. The MPC is backfilled with helium to the design-basis pressures (Table 4.4.8). This provides a stable, inert environment for long-term storage of the SNF. Heat is rejected from the SNF in the HI-STORM FW system to the environment by passive heat transport mechanisms only.

The helium backfill gas plays an important role in the MPC's thermal performance. 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. Within the MPC the pressurized helium environment sustains a closed loop thermosiphon action, removing SNF heat by an upward flow of helium through the storage cells. This MPC internal convection heat dissipation mechanism is illustrated in Figure 4.1.2. On the outside of the MPC a ducted overpack construction with a vertical annulus facilitates an upward flow of air by buoyancy forces. The annulus ventilation flow cools the hot MPC surfaces and safely transports heat to the outside environment. The annulus ventilation cooling mechanism is illustrated in Figure 4.1.1. To ensure that the helium gas is retained and is not diluted by lower conductivity air, the MPC Confinement Boundary is designed as an all-seal-welded pressure vessel with redundant closures. It is demonstrated in Section 12.1 that the failure of one field-welded pressure boundary seal will not result in a breach of the pressure boundary. The helium gas is therefore assumed to be retained in an undiluted state, and is credited in the thermal analyses.

An important thermal design criterion imposed on the HI-STORM FW system is to limit the maximum fuel cladding temperature as well as the fuel basket temperature to within design basis limits for long-term storage of design basis SNF assemblies. An equally important requirement is to minimize temperature gradients in the MPC so as to minimize thermal stresses. In order to meet these design objectives, the MPC baskets are designed to possess certain distinctive characteristics, as summarized below.

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 egg-crate basket structure. The MPC design incorporates top and bottom plenums with interconnected downcomer paths formed by the annulus gap in the aluminum shims. The top plenum is formed by the gap between the bottom of the MPC lid and the top of the honeycomb fuel basket. The bottom plenum is formed by flow holes near the base of all cell walls. The MPC basket is designed to minimize structural discontinuities (i.e., gaps) which introduce added 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 that permits unrestrained axial and radial growth of the basket. The possibility of

stresses due to restraint on basket periphery thermal growth is eliminated by providing adequate basket-to-canister shell gaps to allow for basket thermal growth during all operational modes.

[

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The MPCs regionalized fuel storage scenarios are defined in Figures 1.2.1 and 1.2.2 in Chapter 1 and design maximum decay heat loads for storage of zircaloy clad fuel are listed in Tables 1.2.3 and 1.2.4. The axial heat distribution in each fuel assembly is conservatively assumed to be non-uniformly distributed with peaking in the active fuel mid-height region (see axial burnup profiles in Figures 2.1.3 and 2.1.4). Table 4.1.1 summarizes the principal operating parameters of the HI-STORM FW system.

The fuel cladding temperature limits that the HI-STORM FW system is required to meet are discussed in Section 4.3 and given in Table 2.2.3. Additionally, when the MPCs are deployed for storing High Burnup Fuel (HBF) further restrictions during certain fuel loading operations (vacuum drying) are set forth herein to preclude fuel temperatures from exceeding the normal temperature limits. To ensure explicit compliance, a specific term “short-term operations” is defined in Chapter 2 to cover all fuel loading activities. ISG-11 fuel cladding temperature limits are applied for short-term operations.

The HI-STORM FW system (i.e., HI-STORM FW overpack, HI-TRAC VW transfer cask and MPC) is evaluated under normal storage (HI-STORM FW overpack), during off-normal and accident events and during short-term operations in a HI-TRAC VW. Results of HI-STORM FW thermal analysis during normal (long-term) storage are obtained and reported in Section 4.4. Results of HI-TRAC VW short-term operations (fuel loading, on-site transfer and vacuum drying) are reported in Section 4.5. Results of off-normal and accident events are reported in Section 4.6.

Table 4.1.1	
HI-STORM FW OPERATING CONDITION PARAMETERS	
Condition	Value
MPC Decay Heat, max.	Tables 1.2.3 and 1.2.4
MPC Operating Pressure	Note 1
Normal Ambient Temperature	Table 2.2.2
Helium Backfill Pressure	Table 4.4.8
Note 1: The MPC operating pressure used in the thermal analysis is based on the minimum helium backfill pressure specified in Table 4.4.8 and MPC cavity average temperature.	

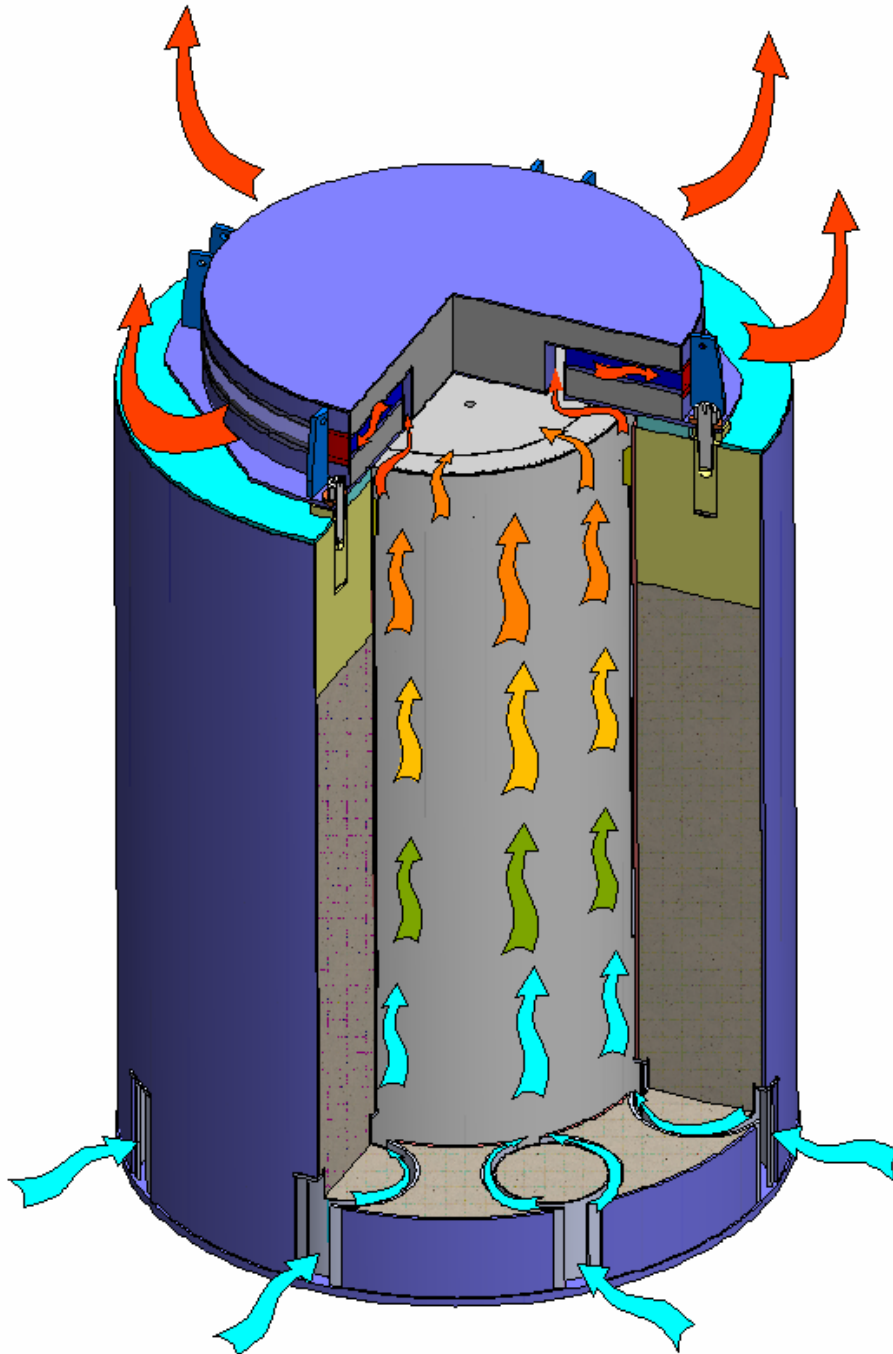


Figure 4.1.1: Ventilation Flow in the HI-STORM FW System

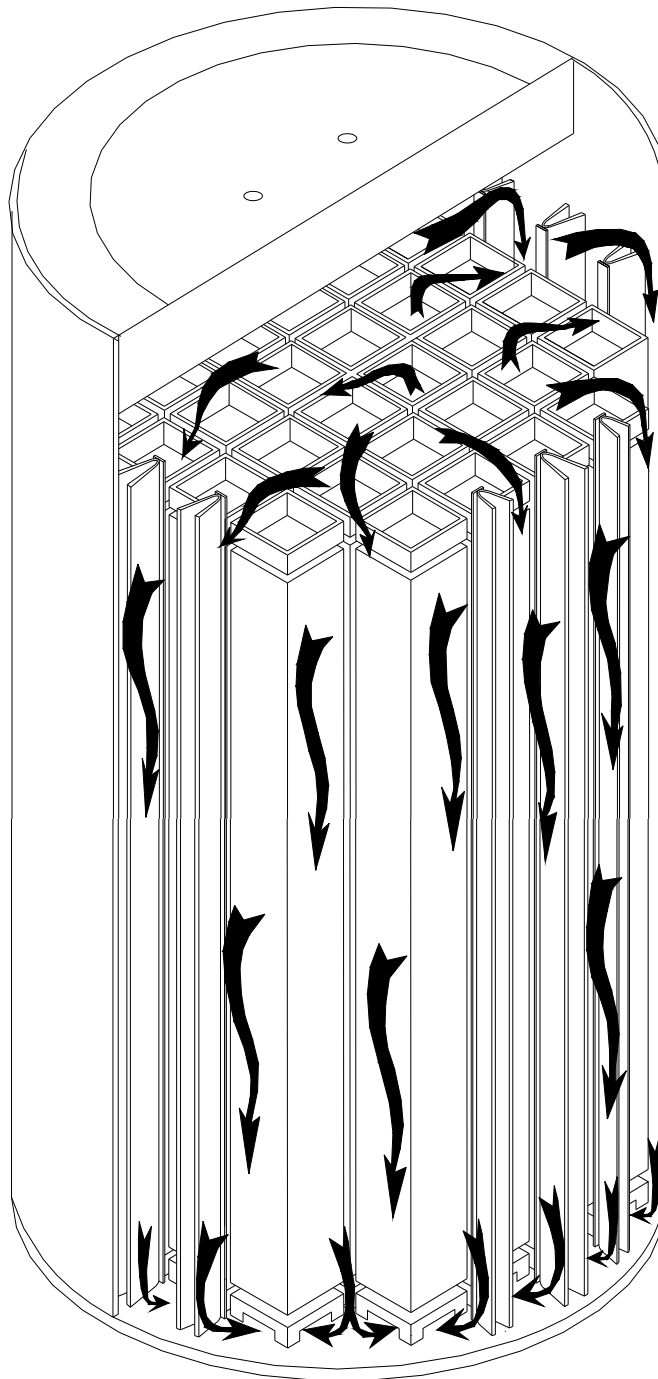


Figure 4.1.2: Illustration of MPC Internal Helium Circulation



## 4.2 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

The thermo-physical properties listed in the tables in this section are identical to those used in the HI-STORM 100 FSAR [4.1.8], except for Metamic-HT and aluminum shims. Materials present in the MPCs include Alloy X<sup>1</sup>, Metamic-HT, aluminum alloy 2219, and helium. Materials present in the HI-STORM FW storage overpack include carbon steels and concrete. Materials present in the HI-TRAC VW transfer cask include carbon steel, lead, air, and demineralized water. In Table 4.2.1, a summary of references used to obtain cask material properties for performing all thermal analyses is presented.

Individual thermal conductivities of the alloys that comprise the Alloy X materials and the bounding Alloy X thermal conductivity are reported in Appendix 1.A of this report. Tables 4.2.2 and 4.2.3 provide numerical thermal conductivity data of materials at several representative temperatures.

Surface emissivity data for key materials of construction are provided in Table 4.2.4. The emissivity properties of painted external 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-STORM FW thermal analysis, an emissivity of 0.85<sup>2</sup> is applied to painted surfaces. The solar absorptivity,  $\alpha_s$  of paints are generally low. The NASA technical publication [4.2.20] reports  $\alpha_s$  in the range of 0.03 to 0.54. For a robustly bounding analysis  $\alpha_s$  equal to 0.85 is applied to all exposed overpack surfaces.

In Table 4.2.5, the heat capacity and density of the MPC, overpack and CSF materials are presented. These properties are used in performing transient (i.e., hypothetical fire accident condition) analyses. The temperature-dependent values of the viscosities of helium and air are provided in Table 4.2.6.

The heat transfer coefficient for exposed surfaces is calculated by accounting for both natural convection and thermal radiation heat transfer. 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  $Gr \times Pr$  is expressed as  $L^3 \Delta T Z$ , where L is height of the overpack,  $\Delta T$  is overpack surface temperature differential and Z is a parameter based on air properties, which are known functions of temperature, evaluated at the average film temperature. The temperature dependent values of Z are provided in Table 4.2.7.

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<sup>1</sup> Alloy X is defined in Appendix 1.A to designate a group of stainless steel alloys permitted for use in the HI-STORM FW system. In this chapter the terms Alloy X and stainless steel are used interchangeably.

<sup>2</sup> This is conservative with respect to prior cask industry practice, which has historically utilized higher emissivities [4.2.16].

Table 4.2.1				
SUMMARY OF HI-STORM FW 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	[4.2.3], [4.2.17], [4.2.18], [4.2.7]	NUREG [4.2.17]	Rust [4.2.4]	Rust [4.2.4]
UO <sub>2</sub>	Note 1	NUREG [4.2.17]	Rust [4.2.4]	Rust [4.2.4]
Stainless Steel (machined forgings) <sup>Note 2</sup>	Kern [4.2.5]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Stainless Steel Plates <sup>Note 3</sup>	ORNL [4.2.11], [4.2.12]	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]
Concrete	Note 1	Marks' [4.2.1]	Appendix 1.D of HI-STORM 100 FSAR [4.1.8]	Handbook [4.2.2]
Lead	Note 1	Handbook [4.2.2]	Handbook [4.2.2]	Handbook [4.2.2]
Water	Note 1	ASME [4.2.10]	ASME [4.2.10]	ASME [4.2.10]
Metamic-HT	Test Data Table 1.2.8	Test Data Table 1.2.8	Test Data Table 1.2.8	Test Data Table 1.2.8
Aluminum Alloy 2219	Test Data Table 1.2.8	ASM [4.2.19]	ASM [4.2.19]	ASM [4.2.19]
Note 1: Emissivity not reported as radiation heat dissipation from these surfaces is conservatively neglected. Note 2: Used in the MPC lid. Note 3: Used in the MPC shell and baseplate.				

Table 4.2.2				
SUMMARY OF HI-STORM FW SYSTEM MATERIALS THERMAL CONDUCTIVITY DATA				
Material	At 200°F (Btu/ft-hr-°F)	At 450°F (Btu/ft-hr-°F)	At 700°F (Btu/ft-hr-°F)	At 1000°F (Btu/ft-hr-°F)
Helium	0.0976	0.1289	0.1575	0.1890
Air*	0.0173	0.0225	0.0272	0.0336
Alloy X	8.4	9.8	11.0	12.4
Carbon Steel	24.4	23.9	22.4	20.0
Concrete**	1.05	1.05	1.05	1.05
Lead	19.4	17.9	16.9	N/A
Water	0.392	0.368	N/A	N/A
Metamic-HT	Table 1.2.8			
Aluminum Alloy 2219 **	69.3	69.3	69.3	69.3
Aluminum Alloy (Solid Shim Plate)***	86.7	86.7	86.7	86.7
<p>* At lower temperatures, Air conductivity is between 0.0139 Btu/ft-hr-°F at 32°F and 0.0176 Btu/ft-hr-°F at 212°F.</p> <p>** Conservatively assumed to be constant for the entire range of temperatures.</p> <p>*** The optional solid shim aluminum plates discussed in Table 1.2.9 must have the tabulated minimum thermal conductivity.</p>				

Table 4.2.3 <sup>1</sup>			
SUMMARY OF FUEL ELEMENT COMPONENTS THERMAL CONDUCTIVITY DATA			
<b>Zircaloy Cladding</b>		<b>Fuel (UO<sub>2</sub>)</b>	
Temperature (°F)	Conductivity (Btu/ft-hr-°F)	Temperature (°F)	Conductivity (Btu/ft-hr-°F)
392	8.28	100	3.48
572	8.76	448	3.48
752	9.60	570	3.24
932	10.44	793	2.28

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<sup>1</sup> See Table 4.2.1 for cited references.

Table 4.2.4	
SUMMARY OF MATERIALS SURFACE EMISSIVITY DATA*	
Material	Emissivity
Zircaloy	0.80
Painted surfaces	0.85
Stainless steel (machined forgings)	0.36
Stainless Steel Plates	0.587**
Carbon Steel	0.66
Metamic-HT***	Table 1.2.8
Extruded Shims (Aluminum Alloy 2219) <sup>‡</sup>	Table 1.2.9
Solid Shims (Aluminum Alloy) <sup>‡</sup>	Table 1.2.9
<p>* See Table 4.2.1 for cited references.</p> <p>** Lower bound value from the cited references in Table 4.2.1.</p> <p>***[</p> <p style="text-align: center;"><b>Withheld in Accordance with 10 CFR 2.390</b></p>	

Table 4.2.5		
DENSITY AND HEAT CAPACITY PROPERTIES SUMMARY*		
Material	Density (lbm/ft <sup>3</sup> )	Heat Capacity (Btu/lbm-°F)
Helium	(Ideal Gas Law)	1.24
Air	(Ideal Gas Law)	0.24
Zircaloy	409	0.0728
Fuel (UO <sub>2</sub> )	684	0.056
Carbon steel	489	0.1
Stainless steel	501	0.12
Concrete	140**	0.156
Lead	710	0.031
Water	62.4	0.999
Metamic-HT	Table 1.2.8	Table 1.2.8
Aluminum Alloy 2219	177.3	0.207
* See Table 4.2.1 for cited references.		
** Conservatively understated value.		

Table 4.2.6			
GASES VISCOSITY* VARIATION WITH TEMPERATURE			
Temperature (°F)	Helium Viscosity (Micropoise)	Temperature (°F)	Air Viscosity (Micropoise)
167.4	220.5	32.0	172.0
200.3	228.2	70.5	182.4
297.4	250.6	260.3	229.4
346.9	261.8	338.4	246.3
463.0	288.7	567.1	293.0
537.8	299.8	701.6	316.7
737.6	338.8	1078.2	377.6
921.2	373.0	-	-
1126.4	409.3	-	-
* Obtained from Rohsenow and Hartnett [4.2.2].			

Table 4.2.7	
VARIATION OF NATURAL CONVECTION PROPERTIES PARAMETER “Z” FOR AIR WITH TEMPERATURE	
Temperature (°F)	Z (ft <sup>-3</sup> °F <sup>-1</sup> )*
40	2.1×10 <sup>6</sup>
140	9.0×10 <sup>5</sup>
240	4.6×10 <sup>5</sup>
340	2.6×10 <sup>5</sup>
440	1.5×10 <sup>5</sup>
* Obtained from Jakob and Hawkins [4.2.9]	

### 4.3 SPECIFICATIONS FOR COMPONENTS

HI-STORM FW 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) are summarized in Tables 2.2.3. The thermal bases supporting the temperature limits are provided in Table 4.3.1. Long-term integrity of SNF is ensured by the HI-STORM FW system thermal evaluation which demonstrates that fuel cladding temperatures are maintained below design basis limits. [

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Compliance to 10CFR72 requires, in part, identification and evaluation of short-term, off-normal and severe hypothetical accident conditions. The inherent mechanical 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-STORM FW system thermal performance, material temperature limits under normal, short-term operations, and off-normal and accident conditions are provided in Table 2.2.3. Fuel temperature limits mandated by ISG-11 [4.1.4] 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.



Table 4.3.1		
TEMPERATURE LIMITS OF CRITICAL COMPONENTS, °F		
Fuel Cladding (Note 1)		
Condition	MBF	HBF
Normal storage	Table 2.2.3	Table 2.2.3
Short-term operations	Table 2.2.3	Table 2.2.3
Off-normal and Accident conditions	Table 2.2.3	Table 2.2.3
Metamic-HT (Note 2)		
Normal storage	Table 2.2.3	
Short term operations, Off-Normal and Accident conditions	Table 2.2.3	
Aluminum Shims (Note 3)		
Normal storage	Table 2.2.3	
Short term operations, Off-normal and Accident conditions	Table 2.2.3	
HI-TRAC VW Jacket		
Short term operations and off-normal conditions	Table 2.2.3 (Note 4)	
Accident condition	NA (Note 5)	
Notes:		
1. Temperature limits per ISG-11, Rev. 3 [4.1.4].		
2. The B <sub>4</sub> C component in Metamic-HT is a refractory material that is unaffected by high temperature (on the order of 1000°F) and the aluminum component is solid at temperatures in excess of 1000°F.		
3. To preclude melting the temperature limits are set well below the melting temperature of Aluminum Alloys.		
4. Temperature limit is defined by the saturation temperature of water at water jacket design pressure specified in Table 2.2.1.		
5. The jacket water is assumed to be lost under accident conditions.		

## 4.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF STORAGE

The HI-STORM FW Storage System (i.e., HI-STORM FW overpack and MPC) and HI-TRAC VW transfer cask thermal evaluation is performed in accordance with the guidelines of NUREG-1536 [4.4.1] and ISG-11 [4.1.4]. To ensure a high level of confidence in the thermal evaluation, 3-dimensional models of the MPC, HI-STORM FW overpack and HI-TRAC VW transfer cask are constructed to evaluate fuel integrity under normal (long-term storage), off-normal and accident conditions and in the HI-TRAC VW transfer cask under short-term operation and hypothetical accidents. The principal features of the thermal models are described in this section for HI-STORM FW and Section 4.5 for HI-TRAC VW. Thermal analyses results for the long-term storage scenarios are obtained and reported in this section. The evaluation addresses the design basis thermal loadings defined in Chapter 1, Tables 1.2.3 (MPC-37, Patterns A and B) and 1.2.4 (MPC-89). Based on these evaluations the limiting thermal loading condition is defined in Subsection 4.4.4 and adopted for evaluation of on-site transfer in the HI-TRAC (Section 4.5) and off-normal and accident events defined in Section 4.6.

### 4.4.1 Overview of the Thermal Model

As illustrated in the drawings in Section 1.5, the basket is a matrix of interconnected square compartments designed to hold the fuel assemblies in a vertical position under long term storage conditions. The basket is a honeycomb structure of Metamic-HT plates that are slotted and arrayed in an orthogonal configuration to form an integral basket structure. [

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Thermal analysis of the HI-STORM FW System is performed for all heat load scenarios defined in Chapter 1 for regionalized storage (Figures 1.2.1 and 1.2.2). Each fuel assembly is *assumed to be generating heat at the maximum permissible rate (Tables 1.2.3 and 1.2.4)*. While the assumption of limiting heat generation in each storage cell imputes a certain symmetry to the cask thermal problem, it grossly overstates the total heat duty of the system in most cases because it is unlikely that any basket would be loaded with fuel emitting heat at their limiting values in *each* storage cell. Thus, the thermal model for the HI-STORM FW system is inherently conservative for real life applications. Other noteworthy features of the thermal analyses are:

- i. While the rate of heat conduction through metals is a relatively weak function of temperature, radiation heat exchange increases rapidly as the fourth power of absolute temperature.
- ii. Heat generation in the MPC is axially non-uniform due to non-uniform axial burnup profiles 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 lowest values reached at the periphery of the basket.

As noted in Chapter 1 and in Section 3.2, the height of the PWR MPC cavity can vary within a rather large range to accommodate spent nuclear fuel of different lengths. The heat load limits in Table 1.2.3 (PWR MPC) and Table 1.2.4 (BWR MPC) for regionalized storage are, however, fixed regardless of the fuel (and hence MPC cavity) length. Because it is not a priori obvious whether the shortest or the longest fuel case will govern, thermal analyses are performed for the minimum<sup>1</sup>, reference and maximum height MPCs. Table 2.1.1 allows two different fuel assembly lengths under “minimum” category for PWR fuel. Unless specified in this chapter, the term “minimum” or “short” is used for all short fuel assembly arrays except 15x15I short fuel defined in Chapter 2.

#### 4.4.1.1 Description of the 3-D Thermal Model

##### i. Overview

The HI-STORM FW System is equipped with two MPC designs, MPC-37 and MPC-89 engineered to store 37 and 89 PWR and BWR fuel assemblies respectively. The interior of the MPC is a 3-D array of square shaped cells inside an irregularly shaped basket outline confined inside the cylindrical space of the MPC cavity. To ensure an adequate representation of these features, a 3-D geometric model of the MPC is constructed using the FLUENT CFD code pre-processor [4.1.2]. Because the fuel basket is made of a single isotropic material (Metamic-HT), the 3-D thermal model requires no idealizations of the fuel basket structure. However, since it is impractical to model every fuel rod in every stored fuel assembly explicitly, the cross-section bounded by the inside of the storage cell (inside of the fuel channel in the case of BWR MPCs), which surrounds the assemblage of fuel rods and the interstitial helium gas (also called the “rodded region”), is replaced with an “equivalent” square homogeneous section characterized by an effective thermal conductivity. Homogenization of the cell cross-section is discussed under item (ii) below. For thermal-hydraulic simulation, each fuel assembly in its storage cell is represented by an equivalent porous medium. For BWR fuel, the presence of the fuel channel divides the storage cell space into two distinct axial flow regions, namely, the in-channel (rodded) region and the square prismatic annulus region (in the case of PWR fuel this modeling complication does not exist). The methodology to represent the spent fuel storage space as a homogeneous region with equivalent conductivities is identical to that used in the HI-STORM 100 Docket No. 72-1014 [4.1.8].

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<sup>1</sup> Both allowable PWR fuel assembly lengths under “minimum” category as shown in Table 2.1.1 are evaluated in this chapter.

ii. Details of the 3-D Model

The HI-STORM FW fuel basket is modeled in the same manner as the model described in the HI-STAR 180 SAR (NRC Docket No. 71-9325) [4.1.11]. Modeling details are provided in the following:

Fuel Basket 3D Model

The MPC-37 and MPC-89 fuel baskets are essentially an array of square cells within an irregularly shaped basket outline. The fuel basket is confined inside a cylindrical cavity of the MPC shell. Between the fuel basket-to-shell spaces, thick Aluminum basket shims are installed to facilitate heat dissipation. To ensure an adequate representation of the fuel basket a geometrically accurate 3D model of the array of square cells and Metamic-HT plates is constructed using the FLUENT pre-processor. Other than the representation of fuel assemblies inside the storage cell spaces as porous region with effective thermal-hydraulic properties as described in the next paragraph, the 3D model includes an explicit articulation of other canister parts. The basket shims are explicitly modeled in the peripheral spaces. The fuel basket is surrounded by the MPC shell and outfitted with a solid welded lid above and a baseplate below. All of these physical details are explicitly articulated in a quarter-symmetric 3D thermal model of the HI-STORM FW.

Fuel Region Effective Planar Conductivity

In the HI-STORM FW thermal modeling, the cross section bounded by the inside of a PWR storage cell and the channeled area of a BWR storage cell is replaced with an “equivalent” square section characterized by an effective thermal conductivity in the planar and axial directions. Figure 4.4.1 pictorially illustrates this concept. The two conductivities are unequal because while in the planar direction heat dissipation is interrupted by inter-rod gaps; in the axial direction heat is dissipated through a continuous medium (fuel cladding). The equivalent planar conductivity of the storage cell space is obtained using a 2D conduction-radiation model of the bounding PWR and BWR fuel storage scenarios defined in the table below. The fuel geometry, consisting of an array of fuel rods with helium gaps between them residing in a storage cell, is constructed using the ANSYS code [4.1.1] and lowerbound conductivities under the assumed condition of stagnant helium (no-helium-flow-condition) are obtained. In the axial direction, an area-weighted average of the cladding and helium conductivities is computed. Axial heat conduction in the fuel pellets is conservatively ignored.

The effective fuel conductivity is computed under four bounding fuel storage configurations for PWR fueled MPC-37 and one bounding scenario for BWR fueled MPC-89. The fuel storage configurations are defined below:

Storage Scenario	MPC	Fuel
PWR: 15x15I Short Fuel	Minimum Height MPC-37 for 15x15I fuel assembly array	15x15I in Table 2.1.2
PWR: Short Fuel	Minimum Height MPC-37 for all fuel assembly arrays except 15x15I	14x14 Ft. Calhoun

PWR: Standard Fuel	Reference Height MPC-37	W-17x17
PWR: XL Fuel	Maximum Height MPC-37	AP1000
BWR	MPC-89	GE-10x10

The fuel region effective conductivity is defined as the calculated equivalent conductivity of the fuel storage cell due to the combined effect of conduction and radiation heat transfer in the manner of the approach used in the HI-STORM 100 system (Docket No. 72-1014). Because radiation is proportional to the fourth power of absolute temperature, the effective conductivity is a strong function of temperature. The ANSYS finite element model is used to characterize fuel resistance at several representative storage cell temperatures and the effective thermal conductivity as a function of temperature obtained for all storage configurations defined above and tabulated in Table 4.4.1.

#### Heat Rejection from External Surfaces

The exposed surfaces of the HI-STORM FW dissipate heat by radiation and external natural convection heat transfer. Radiation is modeled using classical equations for radiation heat transfer (Rohsenow & Hartnett [4.2.2]). Jakob and Hawkins [4.2.9] recommend the following correlations for natural convection heat transfer to air from heated vertical and horizontal surfaces:

Turbulent range:

$$h = 0.19 (\Delta T)^{1/3} \text{ (Vertical, GrPr} > 10^9 \text{)}$$

$$h = 0.18 (\Delta T)^{1/3} \text{ (Horizontal Cylinder, GrPr} > 10^9 \text{)}$$

(in conventional U.S. units)

Laminar range:

$$h = 0.29 \left( \frac{\Delta T}{L} \right)^{1/4} \text{ (Vertical, GrPr} < 10^9 \text{)}$$

$$h = 0.27 \left( \frac{\Delta T}{D} \right)^{1/4} \text{ (Horizontal Cylinder, GrPr} < 10^9 \text{)}$$

(in conventional U.S. Units)

where  $\Delta T$  is the temperature differential between the cask's exterior surface and ambient air and GrPr is the product of Grashof and Prandtl numbers. During storage conditions, the cask cylinder and top surfaces are cooled by natural convection. The corresponding length scales  $L$  for these surfaces are the cask diameter and length, respectively. As described in Section 4.2, Gr $\times$ Pr can be expressed as  $L^3 \Delta T Z$ , where  $Z$  (from Table 4.2.7) is at least  $2.6 \times 10^5$  at a conservatively high surface temperature of 340°F. Thus the turbulent condition is always satisfied assuming a lowerbound  $L$  (8 ft) and a small  $\Delta T$  ( $\sim 10^\circ\text{F}$ ).

### Determination of Solar Heat Input

The intensity of solar radiation incident on exposed surfaces depends on a number of time varying parameters. 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. In the interest of conservatism, the effects of dust, haze, angle of incidence, latitude, etc. that act to reduce insolation, are neglected.

The insolation energy absorbed by the HI-STORM FW is the product of incident insolation and surface absorptivity. To model insolation heating a reasonably bounding absorptivity equal to 0.85 is incorporated in the thermal models. The HI-STORM FW thermal analysis is based on 12-hour daytime insolation specified in Article 71.71(c) (1) of the Transport Regulations [4.6.1]. During long-term storage, the HI-STORM FW Overpack is cyclically subjected to solar heating during the 12-hour daytime period followed by cooling during the 12-hour nighttime. Due to the large mass of metal and the size of the cask, the dynamic time lag exceeds the 12-hour heating period. Accordingly, the HI-STORM FW model includes insolation on exposed surfaces averaged over a 24-hour time period.

### HI-STORM FW Annulus

The HI-STORM FW is engineered with internal flow passages to facilitate heat dissipation by ventilation action. During fuel storage ambient air is drawn from intake ducts by buoyancy forces generated by the heated column of air in the HI-STORM FW annulus. The upward moving air extracts heat from the MPC external surfaces by convection heat transfer. As great bulk of the heat is removed by the annulus air, the adequacy of the grid deployed to model annulus heat transfer must be confirmed prior to performing design basis calculations. To this end a grid sensitivity study is conducted in Subsection 4.4.1.6 to define the converged grid discretization of the annulus region. The converged grid is deployed to evaluate the thermal state of the HI-STORM FW system under normal, off-normal and accident conditions of storage.

### iii. Principal Attributes of the 3D Model

The 3-D model implemented to analyze the HI-STORM FW system is entirely based on the HI-STORM 100 thermal model except that the radiation effect is simulated by the more precise “DO” model (in lieu of the DTRM model used in HI-STORM 100) in FLUENT in the manner of HI-STAR 180 in docket 71-9325. This model has the following key attributes:

- a) The fuel storage spaces are modeled as porous media having effective thermal-hydraulic properties.
- b) In the case of BWR MPC-89, the fuel bundle and the small surrounding spaces inside the fuel “channel” are replaced by an equivalent porous media having the flow impedance properties computed using a conservatively articulated 3-D CFD model [4.4.2]. The space between the BWR fuel channel and the storage cell is represented as an open flow annulus. The fuel channel is also explicitly modeled.

The channeled space within is also referred to as the “rodded region” that is modeled as a porous medium. The fuel assembly is assumed to be positioned coaxially with respect to its storage cell. The MPC-89 storage cell occupied with channeled BWR fuel is shown in Figure 4.4.4.

In the case of the PWR CSF, the porous medium extends to the entire cross-section of the storage cell. As described in [4.4.2], the CFD models for both the BWR and PWR storage geometries are constructed for the Design Basis fuel defined in Table 2.1.4. The model contains comprehensive details of the fuel which includes grid straps, BWR water rods and PWR guide and instrument tubes (assumed to be plugged for conservatism).

- c) The effective conductivities of the MPC storage spaces are computed for bounding fuel storage configurations defined in Paragraph 4.4.1.1(ii). The in-plane thermal conductivities are obtained using ANSYS [4.1.1] finite element models of an array of fuel rods enclosed by a square box. Radiation heat transfer from solid surfaces (cladding and box walls) is enabled in these models. Using these models the effective conduction-radiation conductivities are obtained and reported in Table 4.4.1. For heat transfer in the axial direction an area weighted mean of cladding and helium conductivities are computed (see Table 4.4.1). Axial conduction heat transfer in the fuel pellets and radiation heat dissipation in the axial direction are conservatively ignored. Thus, the thermal conductivity of the rodded region, like the porous media simulation for helium flow, is represented by a 3-D continuum having effective planar and axial conductivities. In the interest of conservatism, thermal analysis of normal storage condition in HI-STORM FW and normal onsite transfer condition in HI-TRAC VW (Section 4.5) are performed with a 10% reduced effective thermal conductivity of fuel region.
- d) The internals of the MPC, including the basket cross-section, aluminum shims, bottom flow holes, top plenum, and circumferentially irregular downcomer formed by the annulus gap in the aluminum shims are modeled explicitly. For simplicity, the flow holes are modeled as rectangular openings with an understated flow area.
- e) The inlet and outlet vents in the HI-STORM FW overpack are modeled explicitly to incorporate any effects of non-axisymmetry of inlet air passages on the system’s thermal performance.
- f) The air flow in the HI-STORM FW/MPC annulus is simulated by the  $k-\omega$  turbulence model with the transitional option enabled. The adequacy of this turbulence model is confirmed in the Holtec benchmarking report [4.1.6]. The annulus grid size is selected to ensure a converged solution.(See Section 4.4.1.6).

- g) A limited number of fuel assemblies (upto 12 in MPC-37 and upto 16 in MPC-89) classified as damaged fuel are permitted to be stored in the MPC inside Damaged Fuel Containers (DFCs). A DFC can be stored in the outer peripheral locations of both MPC-37 and MPC-89 as shown in Figures 2.1.1 and 2.1.2, respectively. DFC emplaced fuel assemblies have a higher resistance to helium flow because of the debris screens. However, DFC fuel storage does not affect temperature of hot fuel stored in the core of the basket because DFC storage is limited by Technical Specifications for placement in the peripheral storage locations away from hot fuel. For this reason the thermal modeling of the fuel basket under the assumption of all storage spaces populated with intact fuel is justified.
- h) As shown in HI-STORM FW drawings in Section 1.5 the HI-STORM FW overpack is equipped with an optional heat shield to protect the inner shell and concrete from radiation heating by the emplaced MPC. The inner and outer shells and concrete are explicitly modeled. All the licensing basis thermal analyses explicitly include the heat shields. A sensitivity study is performed as described in paragraph 4.4.1.9 to evaluate the absence of heat shield on the overpack inner shell and overpack lid.
- i) To maximize lateral resistance to heat dissipation in the fuel basket, 0.8 mm full length inter- panel gaps are conservatively assumed to exist at all intersections. This approach is identical to that used in the thermal analysis of the HI-STAR 180 Package in Docket 71-9325. The shims installed in the MPC peripheral spaces (See MPC-37 and MPC-89 drawings in Section 1.5) are explicitly modeled. For conservatism bounding as-built gaps (3 mm basket-to-shims and 3 mm shims-to-shell) are assumed to exist and incorporated in the thermal models.
- j) The thermal models incorporate all modes of heat transfer (conduction, convection and radiation) in a conservative manner.
- k) The Discrete Ordinates (DO) model, previously utilized in the HI-STAR 180 docket (Docket 71-9325), is deployed to compute radiation heat transfer.
- l) Laminar flow conditions are applied in the MPC internal spaces to obtain a lowerbound rate of heat dissipation.

The 3-D model described above is illustrated in the cross-section for the MPC-89 and MPC-37 in Figures 4.4.2 and 4.4.3, respectively. A closeup of the fuel cell spaces which explicitly include the channel-to-cell gap in the 3-D model applicable to BWR fueled basket (MPC-89) is shown in Figure 4.4.4. The principal 3-D modeling conservatisms are listed below:

- 1) The storage cell spaces are loaded with high flow resistance design basis fuel assemblies (See Table 2.1.4).



- 2) Each storage cell is generating heat at its limiting value under the regionalized storage scenarios defined in Chapter 2, Section 2.1.
- 3) Axial dissipation of heat by conduction in the fuel pellets is neglected.
- 4) Dissipation of heat from the fuel rods by radiation in the axial direction is neglected.
- 5) The fuel assembly channel length for BWR fuel is overstated.
- 6) The most severe environmental factors for long-term normal storage - ambient temperature of 80°F and 10CFR71 insolation levels - were coincidentally imposed on the system.
- 7) Reasonably bounding solar absorptivity of HI-STORM FW overpack external surfaces is applied to the thermal models.
- 8) To understate MPC internal convection heat transfer, the helium pressure is understated.
- 9) No credit is taken for contact between fuel assemblies and the MPC basket wall or between the MPC basket and the basket supports.
- 10) Heat dissipation by fuel basket peripheral supports is neglected.
- 11) Lowerbound fuel basket emissivity function defined in the Metamic-HT Sourcebook [4.2.6] is adopted in the thermal analysis.
- 12) Lowerbound stainless steel emissivity obtained from cited references (See Table 4.2.1) are applied to MPC shell.
- 13) The  $k-\omega$  model used for simulating the HI-STORM FW annulus flow yields uniformly conservative results [4.1.6].
- 14) Fuel assembly length is conservatively modeled equal to the height of the fuel basket.

The effect of crud resistance on fuel cladding surfaces has been evaluated and found to be negligible [4.1.8]. The evaluation assumes a thick crud layer (130  $\mu\text{m}$ ) with a bounding low conductivity (conductivity of helium). The crud resistance increases the clad temperature by a very small amount ( $\sim 0.1^\circ\text{F}$ ) [4.1.8]. Accordingly this effect is neglected in the thermal evaluations.

#### 4.4.1.2 Fuel Assembly 3-Zone Flow Resistance Model<sup>1</sup>

The HI-STORM FW System is evaluated for storage of representative PWR and BWR fuel assemblies determined by a separate analysis, to provide maximum resistance to the axial flow of helium. These are (i) PWR fuel: W17x17 and (ii) BWR fuel: GE10x10. During fuel storage helium enters the MPC fuel cells from the bottom plenum and flows upwards through the open spaces in the fuel storage cells and exits in the top plenum. Because of the low flow velocities the helium flow in the fuel storage cells and MPC spaces is in the laminar regime ( $\text{Re} < 100$ ). The bottom and top plenums are essentially open spaces engineered in the fuel basket ends to facilitate helium circulation. In the case of BWR fuel storage, a channel enveloping the fuel bundle divides the flow in two parallel paths. One flow path is through the in-channel or rodded region of the storage cell and the other flow path is in the square annulus area outside the

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<sup>1</sup> This Sub- section duplicates the methodology used in the HI-STORM FSAR, Rev. 7, supporting CoC Amendment # 5 in Docket 72-1014 [4.1.8].

channel. In the global thermal modeling of the HI-STORM FW System the following approach is adopted:

- (i) In BWR fueled MPCs, an explicit channel-to-cell gap is modeled.
- (ii) The fuel assembly enclosed in a square envelope (fuel channel for BWR fuel or fuel storage cell for PWR fuel) is replaced by porous media with equivalent flow resistance.

The above modeling approach is illustrated in Figure 4.4.4.

In the FLUENT program, porous media flow resistance is modeled as follows:

$$\Delta P/L = D\mu V \quad (\text{Eq. 1})$$

where  $\Delta P/L$  is the hydraulic pressure loss per unit length,  $D$  is the flow resistance coefficient,  $\mu$  is the fluid viscosity and  $V$  is the superficial fluid velocity. In the HI-STORM FW thermal models the fuel storage cell length between the bottom and top plenums<sup>1</sup> is replaced by porous media. As discussed below the porous media length is partitioned in three zones with discrete flow resistances.

To characterize the flow resistance of fuel assemblies inside square envelopes (fuel channel for BWR fuel or fuel storage cell for PWR fuel) 3D models of W-17x17 and GE-10x10 fuel assemblies are constructed using the FLUENT CFD program. These models are embedded with several pessimistic assumptions to overstate flow resistance. These are:

- (a) Water rods (BWR fuel) and guide tubes (PWR fuel) are assumed to be blocked
- (b) Fuel rods assumed to be full length
- (c) Channel length (BWR fuel) overstated
- (d) Bounding grid thickness used
- (e) Bottom fittings resistance overstated
- (f) Bottom nozzle lateral flow holes (BWR fuel) assumed to be blocked

The flow resistance coefficients computed from the 3D flow models [4.4.2] are adopted herein for an MPC-89. In the interest of conservatism, a flow resistance of  $1 \times 10^6 \text{ m}^{-2}$  adopted for thermal hydraulic analysis in Docket 72-1014 CoC amendment 9 is used for PWR fuel assemblies.

#### 4.4.1.3 Bounding Flow Resistance Data

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<sup>1</sup> These are the flow hole openings at the lower end of the fuel basket and a top axial gap to facilitate helium circulation. The flow holes are explicitly included in the 3D thermal models with an understated flow area.

Holtec report [4.4.2] has identified W17x17 OFA and GE 12/14 10x10 fuel assemblies as the design basis fuel for computing the flow resistance coefficients required to compute the in-cell flow of helium in PWR storage cells and of in-channel flow of channeled BWR assemblies placed in a BWR storage cell (See Figure 4.4.4). These resistance coefficients form the basis for the thermal-hydraulic analyses in Docket 72-1014 in the CoC amendments 5. These resistance coefficients are appropriate and conservative for HI-STORM FW analysis because of the following reasons:

- i. The coefficients define the upperbound pressure drop per unit length of fueled space (Eq. 1 in Section 4.4.1.2).
- ii. The storage cell opening in the MPC-37 (PWR fuel) is equal to or greater than the cell openings of the PWR MPCs (such as MPC-32) licensed in the HI-STORM 100 System in Docket 72-1014 [4.1.8]. In the case of BWR fuel storage the channeled fuel located inside the storage cell is modeled explicitly as shown in Figure 4.4.4. The bounding flow resistance coefficients obtained from the cited reference above is applied to the channeled space porous media.
- iii. The length of porous media incorporated in the HI-STORM FW FLUENT models meets or exceeds the fuel assembly length of the longest fuel listed in this SAR.

Thus the flow resistance defined in the manner above is significantly conservative for modeling the Ft. Calhoun 14x14 fuel placed in the limiting minimum height MPC-37 (See Table 4.4.2). The flow resistance for 15x15I short fuel is discussed in Section 4.4.1.7. In the following, explicit calculations for the case of MPC-37 are performed to quantify the conservatism introduced by using the “bounding” resistance data in the FLUENT analysis.

#### 4.4.1.4 Evaluation of Flow Resistance in Enlarged Cell MPCs

The flow resistance factors used in the porous media model are bounding for all fuel types and MPC baskets. This was accomplished for the PWR fueled MPC-37 by placing the most resistive Westinghouse 17x17 fuel assembly in the smaller cell opening MPC-32 approved under the HI-STORM 100 Docket 72-1014, CoC Amendment No. 5 and computing the flow resistance factors. In the case of BWR fueled MPC-89 the most resistive GE-10x10 fuel assembly in the channeled configuration is explicitly modeled in the MPC-89 fuel storage spaces as shown in Figure 4.4.4. The channeled space occupied by the GE-10x10 fuel assembly is modeled as a porous region with effective flow resistance properties computed by deploying an independent 3D FLUENT model of the array of fuel rods and grid spacers.

In the PWR fuel resistance modeling case physical reasoning suggests that the flow resistance of a fuel assembly placed in the larger MPC-37 storage cell will be less than that computed using the (smaller) counterpart cells cavities in the MPC-32. However to provide numerical substantiation FLUENT calculations are performed for the case of W-17x17 fuel placed inside the MPC-32 cell opening of 8.79” and the enlarged MPC-37 cell opening of 8.94”. The FLUENT results for the cell pressure drops under the baseline (MPC-32) and enlarged cell opening (MPC-37) scenarios are shown plotted in Figure 4-4-7. The plot shows that, as expected, the larger cell

cross section case (MPC-37) yields a smaller pressure loss. Therefore, the MPC-37 flow resistance is bounded by the MPC-32 flow resistance used in the FLUENT simulations in the SAR. This evaluation is significant because the MPC-37 basket is determined as the limiting MPC and therefore the licensing basis HI-STORM FW temperatures by use of higher-than-actual resistance are overstated.

However, as mentioned in Sub-section 4.4.1.2, a flow resistance of  $1 \times 10^6 \text{ m}^{-2}$  through PWR fuel assemblies is used in the thermal analysis.

#### 4.4.1.5 Screening Calculations to Ascertain Limiting Storage Scenario

To define the thermally most limiting HI-STORM FW storage scenario the following cases are evaluated under the limiting heat load patterns defined in Tables 1.2.31 and 1.2.4:

- (i) MPC-89
- (ii) Minimum height MPC-37
- (iii) Reference height MPC-37
- (iv) Maximum height MPC-37

To evaluate the above scenarios, 3D FLUENT screening models of the HI-STORM FW cask are constructed, Peak Cladding Temperatures (PCT) computed and tabulated in Table 4.4.2. The results of the calculations yield the following:

- (a) Fuel storage in MPC-37 produces a higher peak cladding temperature than that in MPC-89
- (b) Fuel storage in the minimum height MPC-37 is limiting (produces the highest peak cladding temperature).

To bound the HI-STORM FW storage temperatures the limiting scenario ascertained above is adopted for evaluation of all normal, off-normal and accident conditions.

#### 4.4.1.6 Grid Sensitivity Studies

To achieve grid independent CFD results, a grid sensitivity study is performed on the HI-STORM FW thermal model. The grid refinement is performed in the entire domain i.e. for both fluid and solid regions in both axial and radial directions. Non-uniform meshes with grid cells clustered near the wall regions are generated to resolve the boundary flow near the walls.

A number of grids are generated to study the effect of mesh refinement on the fuel and component temperatures. All sensitivity analyses were carried out for the case of MPC-37 with minimum fuel length under the bounding heat load pattern A. Following table gives a brief summary of the different sets of grids evaluated and PCT results.

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<sup>1</sup> Pattern A defined in Table 1.2.3 is the limiting fuel storage pattern (See Subsection 4.4.4.1).

Mesh No	Total Mesh Size	PCT (°C)	Permissible Limit (°C)	Clad Temperature Margin (°C)
1 (Licensing Basis Mesh)	1,536,882	373	400	27
2	3,354,908	372	400	28
3	7,315,556	372	400	28
Note: Because the flow field in the annulus between MPC shell and overpack inner shell is in the transitional turbulent regime, the value of $y^+$ at the wall-adjacent cell is maintained on the order of 1 to ensure the adequate level of mesh refinement is reached to resolve the viscosity affected region near the wall.				

As can be seen from the above table, the PCT is essentially the same for all the meshes. The solutions from the different grids used are in the asymptotic range. Therefore, it can be concluded that the Mesh 1 is reasonably converged. To provide further assurance of convergence, the sensitivity results are evaluated in accordance with the ASME V&V 20-2009 [4.4.3]. Towards this end, the Grid Convergence Index (GCI), which is a measure of the solution uncertainty, is computed to be 0.181% for these meshes. The apparent order of the method calculated as 2.1, is similar to the order of the method.

Based on the above results, Run No 1 grid layout is adopted for the thermal analysis of the HI-STORM FW.

#### 4.4.1.7 Evaluation of 15x15I Short Fuel Assembly

##### i) Overview

The various fuel assembly types allowed for storage are discussed in Chapter 2. Table 2.1.1 specifically provides a classification of fuel assembly based on its length – minimum, reference and maximum. Due to the unique design of 15x15I short fuel, the thermal evaluation for 15x15I short fuel is separately discussed in this subsection. All the thermal evaluations for MPC-37 discussed previously in this section are applicable to all fuel assembly arrays/classes except the 15x15I fuel defined in Table 2.1.2. The nominal length of 15x15I short fuel with fuel shim is 150 inches (See Table 2.1.1). Therefore, the corresponding height of the MPC, HI-STORM and HI-TRAC are determined based on 150” fuel length.

##### ii) Evaluation of Flow Resistance through Fuel Assemblies

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**Withheld in accordance with 10 CFR 2.390**

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### iii) Three-Dimensional Thermal Model

The converged mesh solution discussed in Sub-Section 4.4.1.6 is modified to model the 15x15 I short fuel assembly length. The mesh in the planar direction is unchanged while the mesh density in the axial direction is maintained. The following additional changes are made to the thermal model discussed in Sub-section 4.4.1.1 thru 4.4.1.6 to evaluate the HI-STORM FW System with 15x15I fuel assembly class:

- (1) The height of basket is conservatively modeled lower (equal to 149 inches).
- (2) The flow resistance through the fuel assemblies is based on the calculations in this sub-section.
- (3) The effective fuel thermal properties specific to 15x15I fuel assembly are used (see Table 4.4.1)

The results of this evaluation are discussed in Sub-section 4.4.4.

#### **4.4.1.8 Thermal Evaluation of Various Basket Shim Design Options**

To allow flexibility in fabrication, the licensing drawings provide various options to install extruded basket shims between the basket and MPC wall. A summary is provided in Table 1.2.9. From a thermal standpoint, emissivity of shims and the average total gap between the basket and extruded shim and extruded shim and MPC shell are critical to heat transfer from the fuel basket to MPC. As noted in Table 1.2.9, solid shim aluminum plates may be placed between the basket and extruded shim to ensure the criteria on the gap in the basket periphery is met.

The predicted temperatures and pressure under normal operating conditions are bounding when MPC is in the HI-TRAC. Therefore, extensive thermal analysis to address all the allowable design options presented in Table 1.2.9 are performed when the MPC is in the HI-TRAC and discussed in Section 4.5. All the licensing basis thermal evaluations for HI-STORM are performed based on Option 1 in Table 1.2.9 since it results in the most limiting PCT and MPC pressure (see supporting evaluations in Section 4.5).

#### **4.4.1.9 Evaluation of Overpack Heat Shields**

HI-STORM FW overpack is equipped with a heat shield on the overpack inner shell and underneath the overpack lid concrete. They are optional features engineered to protect the overpack body concrete and overpack lid concrete from excessive temperature rise due to radiant heat from the MPC. Absence of the heat shields will have an adverse impact on the overpack temperatures. To quantify the impact, a thermal evaluation is performed for a HI-STORM overpack without the heat shields. The thermal model is exactly the same as the converged mesh discussed above in paragraph 4.4.1.6 except that heat shields are removed from the thermal model. The results of this thermal evaluation are discussed in paragraph 4.4.4.4.

#### **4.4.2 Effect of Neighboring Casks**

HI-STORM FW casks are typically stored on an ISFSI pad in regularly spaced arrays (See Section 1.4, Figures 1.4.1 and 1.4.2). Relative to an isolated HI-STORM FW the heat dissipation from a HI-STORM FW cask placed in an array is somewhat disadvantaged. However, as the analysis in this Sub-section shows, the effect of the neighboring casks on the peak cladding temperature in the “surrounded” cask is insignificant.

(i) Effect of Insolation

The HI-STORM FW casks are subject to insolation heating during daytime hours. Presence of surrounding casks has the salutary effect of partially blocking insolation flux. This effect, results in lower temperatures and in the interest of conservatism is ignored in the analysis.

(ii) Effect of Radiation Blocking

The presence of surrounding casks has the effect of partially blocking radiation heat dissipation from the Overpack cylindrical surfaces. Its effect is evaluated in Sub-section 4.4.2.1.

(iii) Effect of Flow Area Reduction

The presence of surrounding casks have the effect of reducing the access flow area around the casks from an essentially unbounded space around it to certain lateral flow passages defined by the spacing between casks (See Figures 1.4.1 and 1.4.2). A reduction in flow area for ventilated casks is not acceptable if the access area falls below the critical flow area in the ventilation flow passages. The HI-STORM FW critical flow area is reached in the narrow annular passage. The lateral flow passages access flow area defined by the product of minimum gap between casks and cask height is computed below. The calculation uses the lowerbound 180 inch cask pitch defined in Table 1.4.1.

Annulus Area ( $A_{min}$ ):

MPC OD: 75.5 in

Overpack ID: 81 in

$A_{min}$ : 676.0 in<sup>2</sup>

Lateral Access Area ( $A_o$ ):

Cask Pitch: 180 in

Overpack OD: 139 in

Overpack Body Height: 187.25 in

Min. cask spacing: 180 – 139 = 41 in

$A_o$ : 7677.2 in<sup>2</sup>

The above numerical exercise shows that  $A_o \gg A_{min}$  and therefore there is an adequate access area surrounding the interior casks for the ventilation air to feed the inlet ducts..

#### 4.4.2.1 Analytical Evaluation of the Effect of Surrounding Casks

In a rectilinear array of HI-STORM FW casks the unit situated in the center of the grid is evidently hydraulically most disadvantaged, because of potential interference to air intake from surrounding casks. Furthermore, the presence of surrounding casks has the effect of partially blocking radiation heat dissipation from the centrally located cask. This situation is illustrated in Figure 4.4.5. To simulate these effects in a conservative manner, a hypothetical square cavity defined by the tributary area  $A_o$  of cask shown in Figure 4.4.5 is erected around the centrally located HI-STORM FW. The hypothetical square cavity has the following attributes:

1. The hypothetical square cavity (with the subject HI-STORM FW situated coaxially in it) is constructed for the 15 ft minimum cask pitch defined in Section 1.4.1.
2. The cavity walls are impervious to air. In this manner as shown in Figure 4.4.6 lateral access to ambient air is choked.
3. The cavity walls are defined as reflecting surfaces from the inside and insulated from the outside. In this manner lateral dissipation of heat by radiation is blocked.
4. The hypothetical square cavity is open at the top as shown in Figure 4.4.6 to allow ambient air access for ventilation cooling in a conservative manner.

The principal results of the hypothetical square cavity thermal model are tabulated below and compared with the baseline thermal results tabulated in Section 4.4.4.

Model	Peak Clad Temperature (°F)	Margin-to-Limit (°F)
Single Cask Model	703	49
Hypothetical Square Cavity Thermal Model	7021	50
Peak cladding temperature reported for the limiting heat load MPC-37 Pattern A (See Subsection 4.4.4.1)		

The results show that the presence of surrounding casks has essentially no effect on the fuel cladding temperatures (the difference in the results is within the range of numerical round-off). These results are in line with prior thermal evaluations of the effect of surrounding casks in the NRC approved HI-STORM 100 System in Docket 72-1014.

#### 4.4.3 Test Model

The HI-STORM FW thermal analysis is performed on the FLUENT [4.1.2] Computational Fluid Dynamics (CFD) program. To ensure a high degree of confidence in the HI-STORM FW thermal evaluations, the FLUENT code has been benchmarked using data from tests conducted

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<sup>1</sup> The lower computed temperature is an artifact of the use of overstated inlet and outlet loss coefficients in the single cask model. The result supports the conclusion that surrounding casks have essentially no effect on the Peak Cladding Temperatures.



with casks loaded with irradiated SNF ([4.1.3],[4.1.7]). The benchmark work is archived in QA validated Holtec reports ([4.1.5],[4.1.6]). These evaluations show that the FLUENT solutions are conservative in all cases. In view of these considerations, additional experimental verification of the thermal design is not necessary. FLUENT has also been used in all Holtec International Part 71 and Part 72 dockets since 1996.

#### 4.4.4 Maximum and Minimum Temperatures

##### 4.4.4.1 Maximum Temperatures

The 3-D model from the previous subsection is used to determine temperature distributions under long-term normal storage conditions for both MPC-89 and MPC-37. Tables 4.4.2, 4.4.3 and 4.4.5 provide key thermal and pressure results from the FLUENT simulations, respectively. Tables 4.4.12 and 4.4.13 respectively provide the temperature and pressure results from the FLUENT simulation of the 15x15I short fuel assembly height based on the methodology discussed in Sub-Section 4.4.1.7. The peak fuel cladding result in these tables is actually overstated by the fact that the 3-D FLUENT cask model incorporates the effective conductivity of the fuel assembly sub-model. Therefore the FLUENT models report the peak temperature *in the fuel storage cells*. Thus, as the fuel assembly models include the fuel pellets, the FLUENT calculated peak temperatures are actually peak pellet centerline temperatures which bound the peak cladding temperatures with a modest margin.

The following observations can be derived by inspecting the temperature field obtained from the thermal models:

- The fuel cladding temperatures are below the regulatory limit (ISG-11 [4.1.4]) under all regionalized storage scenarios defined in Chapter 1 (Figures 1.2.1 and 1.2.2) and thermal loading scenarios defined in Tables 1.2.3 and 1.2.4.
- The limiting fuel temperatures are reached under the Pattern A thermal loading condition defined in Table 1.2.3 in the MPC-37. Accordingly this scenario is adopted for thermal evaluation under on-site transfer (Section 4.5) and under off-normal and accident conditions (Section 4.6).
- The maximum temperature of the basket structural material is within its design limit.
- The maximum temperatures of the MPC pressure boundary materials are below their design limits.
- The maximum temperatures of concrete are within the guidance of the governing ACI Code (see Table 2.2.3).

- The calculated fuel temperature for the 15x15I short fuel assembly (Table 4.4.12) is bounded by the thermal evaluations for the minimum MPC-37 for short fuel (Table 4.4.3). The temperatures of other cask components are similar. It is reasonable to conclude that the temperatures and pressure for the minimum height MPC-37 (short fuel) bounds all scenarios.

The above observations lead us to conclude that the temperature field in the HI-STORM FW System with a loaded MPC containing heat emitting SNF complies with all regulatory temperature limits (Table 2.2.3). In other words, the thermal environment in the HI-STORM FW System is in compliance with Chapter 2 Design Criteria.

Also, all the licensing basis thermal evaluations documented in this chapter are performed for the most limiting thermal scenarios i.e. minimum MPC-37 with heat load pattern A.

#### 4.4.4.2 Minimum Temperatures

In Table 2.2.2 of this report, the minimum ambient temperature condition for the HI-STORM FW storage overpack and MPC 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 a temperature of -40°F. Low service temperature (-40°F) evaluation of the HI-STORM FW is provided in Chapter 3. All HI-STORM FW storage overpack and MPC materials of construction will satisfactorily perform their intended function in the storage mode under this minimum temperature condition.

#### 4.4.4.3 Effect of Elevation

The reduced ambient pressure at site elevations significantly above the sea level will act to reduce the ventilation air mass flow, resulting in a net elevation of the peak cladding temperature. However, the ambient temperature (i.e., temperature of the feed air entering the overpack) also drops with the increase in elevation. Because the peak cladding temperature also depends on the feed air temperature (the effect is one-for-one within a small range, i.e., 1°F drop in the feed air temperature results in ~1°F drop in the peak cladding temperature), the adverse ambient pressure effect of increased elevation is partially offset by the ambient air temperature decrease. The table below illustrates the variation of air pressure and corresponding ambient temperature as a function of elevation.

Elevation (ft)	Pressure (psia)	Ambient Temperature Reduction versus Sea Level
Sea Level (0)	14.70	0°F
2000	13.66	7.1°F
4000	12.69	14.3°F

A survey of the elevation of nuclear plants in the U.S. shows that nuclear plants are situated near about sea level or elevated slightly (~1000 ft). The effect of the elevation on peak fuel cladding temperatures is evaluated by performing calculations for a HI-STORM FW system situated at an elevation of 1500 feet. At this elevation the ambient temperature would decrease by approximately 5°F (See Table above). The peak cladding temperatures are calculated under the reduced ambient temperature and pressure at 1500 feet elevation for the limiting regionalized storage scenario evaluated in Table 4.4.2. The results are presented in Table 4.4.9.

These results show that the PCT, including the effects of site elevation, continues to be well below the regulatory cladding temperature limit of 752°F. In light of the above evaluation, it is not necessary to place ISFSI elevation constraints for HI-STORM FW deployment at elevations up to 1500 feet. If, however, an ISFSI is sited at an elevation greater than 1500 feet, the effect of altitude on the PCT shall be quantified as part of the 10 CFR 72.212 evaluation for the site using the site ambient conditions.

#### **4.4.4.4 Evaluation of Overpack Heat Shields**

As discussed in Sub-section 4.4.1.9 above, a thermal evaluation is performed to evaluate the effect of removal of heat shields from a HI-STORM overpack. The predicted temperatures from this sensitivity study of normal condition of storage are summarized in Table 4.4.14. The peak cladding temperature, basket and MPC component temperatures decrease due to removal of heat shields. As expected, the results demonstrate an increase in overpack component temperatures. However, the overpack component temperatures are below their respective normal temperature limits with significant margins. Therefore, removal of heat shields does not have a detrimental effect on the system's thermal performance.

The temperatures of overpack components increase due to removal of heat shields under normal conditions of storage. This temperature increase is then added to the predicted temperatures of all the off-normal and accident conditions discussed in Section 4.6. The resulting temperatures are still well below their respective temperature limits which demonstrate that safety conclusions made for all the off-normal and accident condition evaluations in Section 4.6 remain valid even after the removal of heat shields from the HI-STORM overpack.

#### **4.4.5 Maximum Internal Pressure**

##### **4.4.5.1 MPC Helium Backfill Pressure**

The quantity of helium emplaced in the MPC cavity shall be sufficient to produce an operating pressure of 7.1 and 7.0 atmospheres (absolute) respectively for loading patterns A and B during normal storage conditions defined in Table 4.1.1. Thermal analyses performed on the different MPC designs indicate that this operating pressure requires a certain minimum helium backfill pressure ( $P_b$ ) specified at a reference temperature (70°F). The minimum backfill pressure for each MPC type is provided in Table 4.4.7. A theoretical upper limit on the helium backfill

pressure also exists and is defined by the design pressure of the MPC vessel (Table 2.2.1). The upper limit of  $P_b$  is also reported in Table 4.4.7. To bound the minimum and maximum backfill pressures listed in Table 4.4.7 with a margin, a helium backfill specification is set forth in Table 4.4.8.

To provide additional helium backfill range for less than design basis heat load canisters a Sub-Design-Basis (SDB) heat load scenario is defined below:

- (i) MPC-37 under 80% Pattern A Heat Load (Table 1.2.3)
- (ii) MPC-37 under 90% Pattern A Heat Load (Table 1.2.3)
- (iii) MPC-89 under 80% Design Heat Load (Table 1.2.4)
- (iv) MPC-37 under vacuum drying threshold heat load in Table 4.5.11.
- (v) MPC-89 under vacuum drying threshold heat load in Table 4.5.1\*.

The storage cell and MPC heat load limits under the SDB scenario (i), (ii) & (iii) are specified in Table 4.4.11. Calculations for bounding scenarios (i), (ii) & (iii) show that the maximum cladding temperature under the SDB scenario meet the ISG-11 temperature limits. The helium backfill pressure limits supporting this scenario are defined in Table 4.4.10. These backfill limits maybe optionally adopted by a cask user if the decay heats of the loaded fuel assemblies meet the SDB decay heat limits stipulated above.

Two methods are available for ensuring that the appropriate quantity of helium has been placed in an MPC:

- i. By pressure measurement
- ii. By measurement of helium backfill volume (in standard cubic feet)

The direct pressure measurement approach is more convenient if the FHD method of MPC drying is used. In this case, a certain quantity of helium is already in the MPC. Because the helium is mixed inside the MPC during the FHD operation, the temperature and pressure of the helium gas at the MPC's exit provides a reliable means to compute the inventory of helium. A shortfall or excess of helium is adjusted by a calculated raising or lowering of the MPC pressure such that the reference MPC backfill pressure is within the range specified in Table 4.4.8 or Table 4.4.10 (as applicable).

When vacuum drying is used as the method for MPC drying, then it is more convenient to fill the MPC by introducing a known quantity of helium (in standard cubic feet) by measuring the quantity of helium introduced using a calibrated mass flow meter or other measuring apparatus. The required quantity of helium is computed by the product of net free volume and helium specific volume at the reference temperature (70°F) and a target pressure that lies in the mid-range of the Table 4.4.8 pressures.

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1 Threshold scenarios (iv) and (v) are bounded by scenarios (i) and (iii) respectively because the core Region 1 assembly heat loads and total cask heat loads are bounded by the Sub-Design Basis heat loads in Table 4.4.11.

The net free volume of the MPC is obtained by subtracting B from A, where

A = MPC cavity volume in the absence of contents (fuel and non-fuel hardware) computed from nominal design dimensions

B = Total volume of the contents (fuel including DFCs, if used) based on nominal design dimensions

Using commercially available mass flow totalizers or other appropriate measuring devices, an MPC cavity is filled with the computed quantity of helium.

#### 4.4.5.2 MPC Pressure Calculations

The MPC is initially filled with dry 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. The gas pressure inside the MPC will also increase with rising temperature. The pressure rise is determined using the ideal gas law. The MPC gas pressure is also subject to substantial pressure rise under hypothetical rupture of fuel rods and large gas inventory non-fuel hardware (PWR BPRAs). To minimize MPC gas pressures the number of BPRA containing fuel assemblies must be limited to 30.

Table 4.4.4 presents a summary of the MPC free volumes determined for the fixed height MPC-89 and lowerbound height MPC-37 fuel storage scenarios. The MPC maximum gas pressure is computed for a postulated release of fission product gases from fuel rods into this free space. For these scenarios, 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. A concomitant effect of rod ruptures is the increased pressure and molecular weight of the cavity gases with enhanced rate of heat dissipation by internal helium convection and lower cavity temperatures. As these effects are substantial under large rod ruptures the 100% rod rupture accident is evaluated with due credit for increased heat dissipation under increased pressure and molecular weight of the cavity gases. Based on fission gases release fractions (NUREG 1536 criteria [4.4.1]), rods' net free volume and initial fill gas pressure, maximum gas pressures with 1% (normal), 10% (off-normal) and 100% (accident condition) rod rupture are given in Table 4.4.5. The results of the calculations support the following conclusions:

- (i) The maximum computed gas pressures reported in Table 4.4.5 under all design basis thermal loadings defined in Section 4.4 are all below the MPC internal design pressures for normal, off-normal and accident conditions specified in Table 2.2.1.
- (ii) The MPC gas pressure obtained under loading Pattern A is essentially same as in Pattern B. Accordingly Pattern A loading condition for pressure boundary evaluation of MPC in the HI-TRAC and under off-normal and accident conditions is retained.

### Evaluation of Non-Fuel Hardware

The inclusion of PWR non-fuel hardware (BPRA control elements and thimble plugs) to the PWR basket influences the MPC internal pressure through two distinct effects. The presence of non-fuel hardware increases the effective basket conductivity, thus enhancing heat dissipation and lowering fuel temperatures as well as the temperature of 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 on the MPC cavity pressure. The first effect lowers gas pressure while the second effect raises it. In the HI-STORM FW thermal analysis, the computed temperature field (with non-fuel hardware excluded) has been determined to provide a conservatively bounding temperature field for the PWR baskets. The MPC cavity free space is computed based on conservatively computed volume displacement by fuel with non-fuel hardware included. This approach ensures conservative bounding pressures.

During in-core irradiation of BPRAs, neutron capture by the B-10 isotope in the neutron absorbing material produces helium. Two different forms of the neutron absorbing material are used in BPRAs: Borosilicate 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 operation conditions (2,300 psia and approximately 600°F 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 temperatures 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. They differ in the number, diameter, and length of poison rods. The older Westinghouse fuel (W-14x14 and W-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 bounds 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 BPRA containing fuel assembly. For a bounding evaluation the maximum permissible number of BPRA containing fuel assemblies (see discussion at the beginning of this Section) are assumed to be loaded. The MPC cavity pressures (including helium from BPRAs) are summarized in Table 4.4.5 for the bounding MPC-37 (minimum MPC height and heat load Patterns A and B) and MPC-89 (design heat load) storage scenarios.

#### **4.4.6 Engineered Clearances to Eliminate Thermal Interferences**

Thermal stress in a structural component is the resultant sum of two factors, namely: (i) restraint of free end expansion and (ii) non-uniform temperature distribution. To minimize thermal

stresses in load bearing members, the HI-STORM FW system is engineered with adequate gaps to permit free thermal expansion of the fuel basket and MPC in axial and radial directions. In this subsection, differential thermal expansion calculations are performed to demonstrate that engineered gaps in the HI-STORM FW System are adequate to accommodate thermal expansion of the fuel basket and MPC.

The HI-STORM FW System is engineered with gaps for the fuel basket and MPC to expand thermally without restraint of free end expansion. The following gaps are evaluated:

- a. Fuel Basket-to-MPC Radial Gap
- b. Fuel Basket-to-MPC Axial Gap
- c. MPC-to-Overpack Radial Gap
- d. MPC-to-Overpack Axial Gap

The FLUENT thermal model provides the 3-D temperature field in the HI-STORM FW system from which the changes in the above gaps are directly computed. Table 4.4.6 provides the initial minimum gaps and their corresponding value during long-term storage conditions. Significant margins against restraint to free-end expansion are available in the design.

#### **4.4.7 Evaluation of System Performance for Normal Conditions of Storage**

The HI-STORM FW System thermal analysis is based on a detailed 3-D heat transfer model that conservatively accounts for all modes of heat transfer in the MPC and overpack. The thermal model incorporates conservative assumptions that render the results for long-term storage to be conservative.

Temperature distribution results obtained from this thermal model show that the maximum fuel cladding temperature limits are met with adequate margins. Expected margins during normal storage will be much greater due to the conservative assumptions incorporated in the analysis. As justified next the long-term impact of elevated temperatures reached in the HI-STORM FW system is minimal. The maximum MPC basket temperatures are below the recommended limits for susceptibility to stress, corrosion and creep-induced degradation. A complete evaluation of all material failure modes and causative mechanisms has been performed in Chapter 8 which shows that all selected materials for use in the HI-STORM FW system will render their intended function for the service life of the system. Furthermore, stresses induced due to the associated temperature gradients are modestly low (See Structural Evaluation Chapter 3).

Table 4.4.1				
EFFECTIVE FUEL PROPERTIES UNDER BOUNDING FUEL STORAGE CONFIGURATIONS <sup>Note 1</sup>				
	Conductivity (Btu/hr-ft-°F)			
	PWR: Short Fuel		PWR: Standard Fuel	
Temperature (°F)	Planar	Axial	Planar	Axial
200	0.247	0.813	0.231	0.759
450	0.443	0.903	0.387	0.845
700	0.730	1.016	0.601	0.951
	PWR: XL Fuel		BWR Fuel	
	Planar	Axial	Planar	Axial
200	0.239	0.787	0.283	0.897
450	0.393	0.875	0.426	0.988
700	0.599	0.984	0.607	1.104
PWR: 15x15I Short Fuel				
Temperature (°F)	Planar		Axial	
200	0.226		0.763	
450	0.386		0.848	
700	0.601		0.955	
Thermal Inertia Properties				
	Density (lb/ft <sup>3</sup> )		Heat Capacity (Btu/lb-°F) <sup>Note 2</sup>	
PWR: 15x15I Short Fuel	194.5		0.056	
PWR: Short Fuel	165.8		0.056	
PWR: Standard Fuel	176.2		0.056	
PWR: XL Fuel	187.5		0.056	
BWR Fuel	255.6		0.056	
Note 1: Bounding fuel storage configurations defined in 4.4.1.1(ii).				
Note 2: The lowerbound heat capacity of principal fuel assembly construction materials tabulated in Table 4.2.5 (UO <sub>2</sub> heat capacity) is conservatively adopted.				
Note 3: The fuel properties tabulated herein are used in screening calculations to define the limiting scenario for fuel storage (See Table 4.4.2).				



Table 4.4.2	
RESULTS OF SCREENING CALCULATIONS UNDER NORMAL STORAGE CONDITIONS	
Storage Scenario	Peak Cladding Temperature, °C (°F)
MPC-37	
Minimum Height <sup>1</sup>	<b>353 (667)</b>
Reference Height	342 (648)
Maximum Height	316 (601)
MPC-89	333 (631)
Notes:	
(1) The highest temperature highlighted above is reached under the case of minimum height MPC-37 designed to store the short height Ft. Calhoun 14x14 fuel. This scenario is adopted in Chapter 4 for the licensing basis evaluation of fuel storage in the HI-STORM FW system.	
(2) All the screening calculations were performed using a reference coarse mesh [4.1.9] and flow resistance based on the calculations in Holtec report [4.4.2].	

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<sup>1</sup> Bounding scenario adopted in this Chapter for all thermal evaluations.

Table 4.4.3	
MAXIMUM TEMPERATURES IN HI-STORM FW UNDER LONG-TERM NORMAL STORAGE <sup>1</sup>	
Component	Temperature, °C (°F) Pattern A / Pattern B
Fuel Cladding	373 (703) / 368 (694)
MPC Basket	358 (676) / 354 (669)
Basket Periphery	290 (554) / 292 (558)
Aluminum Basket Shims	267 (513) / 267 (513)
MPC Shell	240 (464) / 242 (468)
MPC Lid <sup>Note 1</sup>	235 (455) / 232 (450)
Overpack Inner Shell	126 (259) / 127 (261)
Overpack Outer Shell	65 (149) / 65 (149)
Overpack Body Concrete <sup>Note 1</sup>	89 (192) / 90 (194)
Overpack Lid Concrete <sup>Note 1</sup>	111 (232) / 112 (234)
Area Averaged Air outlet <sup>2</sup>	103 (217) / 103 (217)
Note 1: Maximum section average temperature is reported.	

1 The temperatures reported in this table (all for short fuel scenarios of MPC-37) are below the design temperatures specified in Table 2.2.3, Chapter 2. These temperatures bound MPC-89 temperatures.

2 Reported herein for the option of temperature measurement surveillance of outlet ducts air temperature as set forth in the Technical Specifications.

Table 4.4.4		
MINIMUM MPC FREE VOLUMES		
Item	Lowerbound Height MPC-37 (ft <sup>3</sup> )	MPC-89 (ft <sup>3</sup> )
Net Free Volume*	211.89	210.12
*Net free volumes are obtained by subtracting basket, fuel, aluminum shims, spacers, basket supports and DFCs metal volume from the MPC cavity volume.		

Table 4.4.5		
SUMMARY OF MPC INTERNAL PRESSURES UNDER LONG-TERM STORAGE*		
Condition	MPC-37 (psig) Pattern A/Pattern B	MPC-89 (psig)
Initial backfill** (at 70°F)	45.5/46.0	47.5
Normal: intact rods	96.6/97.9	98.4
1% rods rupture	97.7/99.0	99.0
Off-Normal (10% rods rupture)	107.5/108.9	104.0
Accident (100% rods rupture)	191.5/194.4	155.0
<p>* Per NUREG-1536, pressure analyses with ruptured fuel rods (including BPRA rods for PWR fuel) is performed with release of 100% of the ruptured fuel rod fill gas and 30% of the significant radioactive gaseous fission products.</p> <p>** Conservatively assumed at the Tech. Spec. maximum value (see Table 4.4.8).</p>		

Table 4.4.6

SUMMARY OF HI-STORM FW DIFFERENTIAL THERMAL EXPANSIONS			
Gap Description	Cold Gap U (in)	Differential Expansion $\delta_i$ (in)	Is Free Expansion Criterion Satisfied (i.e., $U > \delta_i$ )
Fuel Basket-to-MPC Radial Gap	0.125	0.112	Yes
Fuel Basket-to-MPC Minimum Axial Gap	1.5	0.421	Yes
MPC-to-Overpack Radial Gap	5.5	0.128	Yes
MPC-to-Overpack Minimum Axial Gap	3.5	0.372	Yes

Table 4.4.7

THEORETICAL LIMITS* OF MPC HELIUM BACKFILL PRESSURE**		
MPC	Minimum Backfill Pressure (psig)	Maximum Backfill Pressure (psig)
MPC-37 Pattern A	41.0	47.3
MPC-37 Pattern B	40.8	47.1
MPC-89	41.9	48.4
* The helium backfill pressures are set forth in the Technical Specifications with a margin (see Table 4.4.8).		
** The pressures tabulated herein are at 70°F reference gas temperature.		

Table 4.4.8 MPC HELIUM BACKFILL PRESSURE SPECIFICATIONS		
MPC	Item	Specification
MPC-37 Pattern A	Minimum Pressure	42.0 psig @ 70°F Reference Temperature
	Maximum Pressure	45.5 psig @ 70°F Reference Temperature
MPC-37 Pattern B	Minimum Pressure	41.0 psig @ 70°F Reference Temperature
	Maximum Pressure	46.0 psig @ 70°F Reference Temperature
MPC-89	Minimum Pressure	42.5 psig @ 70°F Reference Temperature
	Maximum Pressure	47.5 psig @ 70°F Reference Temperature

Table 4.4.9 MAXIMUM HI-STORM FW TEMPERATURES AT ELEVATED SITES <sup>1</sup>	
Component	Temperature, °C (°F)
Fuel Cladding	374 (705)
MPC Basket	360 (680)
Aluminum Basket Shims	275 (527)
MPC Shell	246 (475)
MPC Lid <sup>Note 1</sup>	242 (468)
Overpack Inner Shell	126 (259)
Overpack Body Concrete <sup>Note 1</sup>	86 (187)
Overpack Lid Concrete <sup>Note 1</sup>	112 (234)
Note 1: Maximum section average temperature is reported.	

<sup>1</sup> The temperatures reported in this table (all for the bounding scenario defined in Table 4.4.2) are below the design temperatures specified in Table 2.2.3, Chapter 2.

Table 4.4.10  
MPC HELIUM BACKFILL PRESSURE LIMITS UNDER THE  
SUB-DESIGN-BASIS HEAT LOAD SCENARIO<sup>Note 1</sup>

<b>MPC</b>	<b>Item</b>	<b>Specification</b>
MPC-37 80% of Pattern A	Minimum Pressure	42.0 psig @ 70°F Reference Temperature
	Maximum Pressure	50.0 psig @ 70°F Reference Temperature
MPC-37 90% of Pattern A	Minimum Pressure	42.0 psig @ 70°F Reference Temperature
	Maximum Pressure	47.8 psig @ 70°F Reference Temperature
MPC-89 80% of Table 1.2.4	Minimum Pressure	42.0 psig @ 70°F Reference Temperature
	Maximum Pressure	50.0 psig @ 70°F Reference Temperature
MPC-37 Table 4.5.1 Threshold Heat Load	Minimum Pressure	42.0 psig @ 70°F Reference Temperature
	Maximum Pressure	50.0 psig @ 70°F Reference Temperature
MPC-89 Table 4.5.1 Threshold Heat Load	Minimum Pressure	42.0 psig @ 70°F Reference Temperature
	Maximum Pressure	50.0 psig @ 70°F Reference Temperature

Note 1: The Sub-Design-Basis heat load scenario is defined in Section 4.4.5.1.

Note 2: Sub-design-basis heat load MPCs are sufficiently backfilled to yield an absolute operating pressure of 6 atm in 80% heat load cases and 6.9 atm in 90% heat load cases.

Note 3: The 80% heat load backfill specifications are suitably adopted for threshold heat load scenarios because the thermal scenarios bound the latter (See Subsection 4.4.5.1).

Table 4.4.11 SUB-DESIGN BASIS HEAT LOAD LIMITS	
<u>MPC-37 (80% of Pattern A in Table 1.2.3)</u> Region 1 Cells Region 2 Cells Region 3 Cells Total	0.840 kW/assy 1.360 kW/assy 0.712 kW/assy 35.27 kW
<u>MPC-37 (90% of Pattern A in Table 1.2.3)</u> Region 1 Cells Region 2 Cells Region 3 Cells Total	0.945 kW/assy 1.530 kW/assy 0.801 kW/assy 39.68 kW
<u>MPC-89 (80% of Table 1.2.4)</u> Region 1 Cells Region 2 Cells Region 3 Cells Total	0.352 kW/assy 0.496 kW/assy 0.352 kW/assy 37.1 kW
Note: The MPC-37 and MPC-89 storage cell regions are defined in Figures 1.2.1 and 1.2.2 respectively.	

Table 4.4.12	
MAXIMUM TEMPERATURES IN HI-STORM FW UNDER LONG-TERM NORMAL STORAGE FOR 15x15I SHORT FUEL ASSEMBLY LENGTH <sup>1</sup>	
Component	Temperature, °C (°F) Pattern A
Fuel Cladding	368 (694)
MPC Basket	352 (666)
Basket Periphery	295 (563)
Aluminum Basket Shims	283 (541)
MPC Shell	250 (482)
MPC Lid <sup>Note 1</sup>	247 (477)
Overpack Inner Shell	130 (266)
Overpack Outer Shell	66 (151)
Overpack Body Concrete <sup>Note 1</sup>	91 (196)
Overpack Lid Concrete <sup>Note 1</sup>	113 (235)
Area Averaged Air outlet	104 (219)
Note 1: Maximum section average temperature is reported.	

<sup>1</sup> The temperatures reported in this table are below the design temperatures specified in Table 2.2.3, Chapter 2.



Table 4.4.13	
SUMMARY OF MPC INTERNAL PRESSURES UNDER LONG-TERM STORAGE FOR 15x15I SHORT FUEL ASSEMBLY*	
Condition	Gauge Pressure (psig)
Maximum Initial backfill at 21.1°C (70°F)**	45.5
Normal condition (no rods ruptured)	96.2
Normal condition (1% rods ruptured)***	97.1
Off-normal (10% rods ruptured)***	105.6
Accident (100% rods ruptured)***	190.8
<p>* Per NUREG-1536, pressure analyses with ruptured fuel rods (including BPRA rods for PWR fuel) is performed with release of 100% of the ruptured fuel rod fill gas and 30% of the significant radioactive gaseous fission products.</p> <p>** Conservatively assumed at the Tech. Spec. maximum value (see Table 4.4.8).</p> <p>*** Fuel assembly class 15x15I short fuel defined in Table 2.1.2 do not have BPRAs. Therefore the MPC cavity pressure due to rod ruptures does not include any contribution from BPRAs for this type of fuel assembly.</p>	

Table 4.4.14	
MAXIMUM TEMPERATURES IN HI-STORM FW UNDER LONG-TERM NORMAL STORAGE FOR AN OVERPACK WITHOUT HEAT SHIELDS <sup>1</sup>	
Component	Temperature, °C (°F)
Fuel Cladding	368 (694)
MPC Basket	354 (669)
Basket Periphery	286 (547)
Aluminum Basket Shims	262 (504)
MPC Shell	234 (453)
MPC Lid <sup>Note 1</sup>	225 (437)
Overpack Inner Shell	156 (313)
Overpack Outer Shell	65 (149)
Overpack Body Concrete <sup>Note 1</sup>	104 (219)
Overpack Lid Concrete <sup>Note 1</sup>	106 (223)
Area Averaged Air outlet <sup>2</sup>	97 (207)
Note 1: Maximum section average temperature is reported.	

- 
- 1 The temperatures reported in this table (licensing basis scenario – short fuel in an MPC-37) are below the design temperatures specified in Table 2.2.3, Chapter 2.
- 2 Reported herein for the option of temperature measurement surveillance of outlet ducts air temperature as set forth in the Technical Specifications.

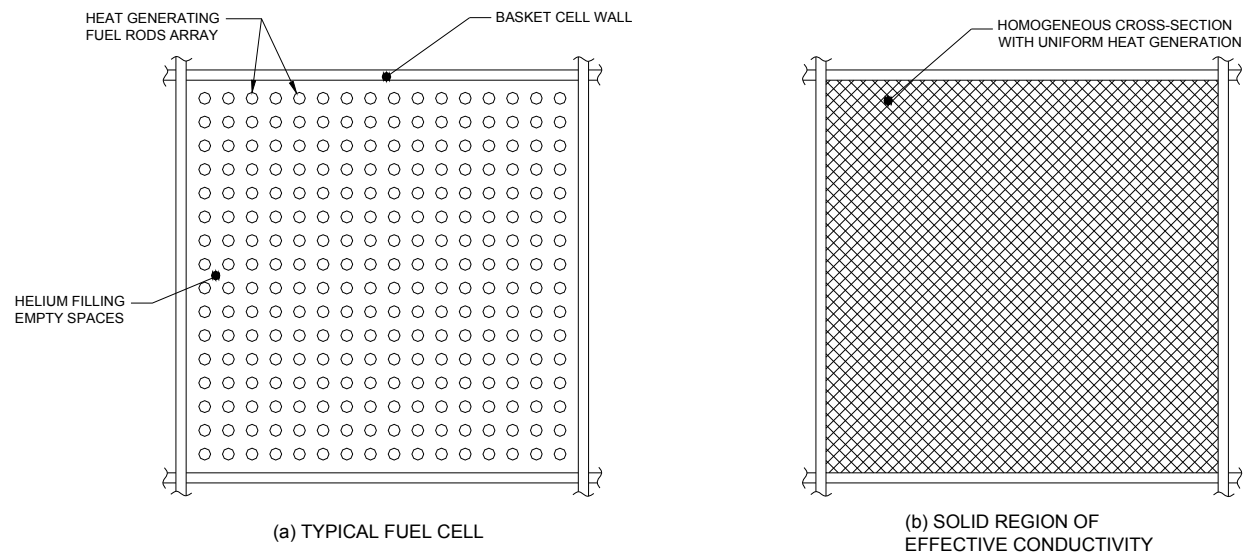


Figure 4.4.1: Homogenization of the Storage Cell Cross-Section

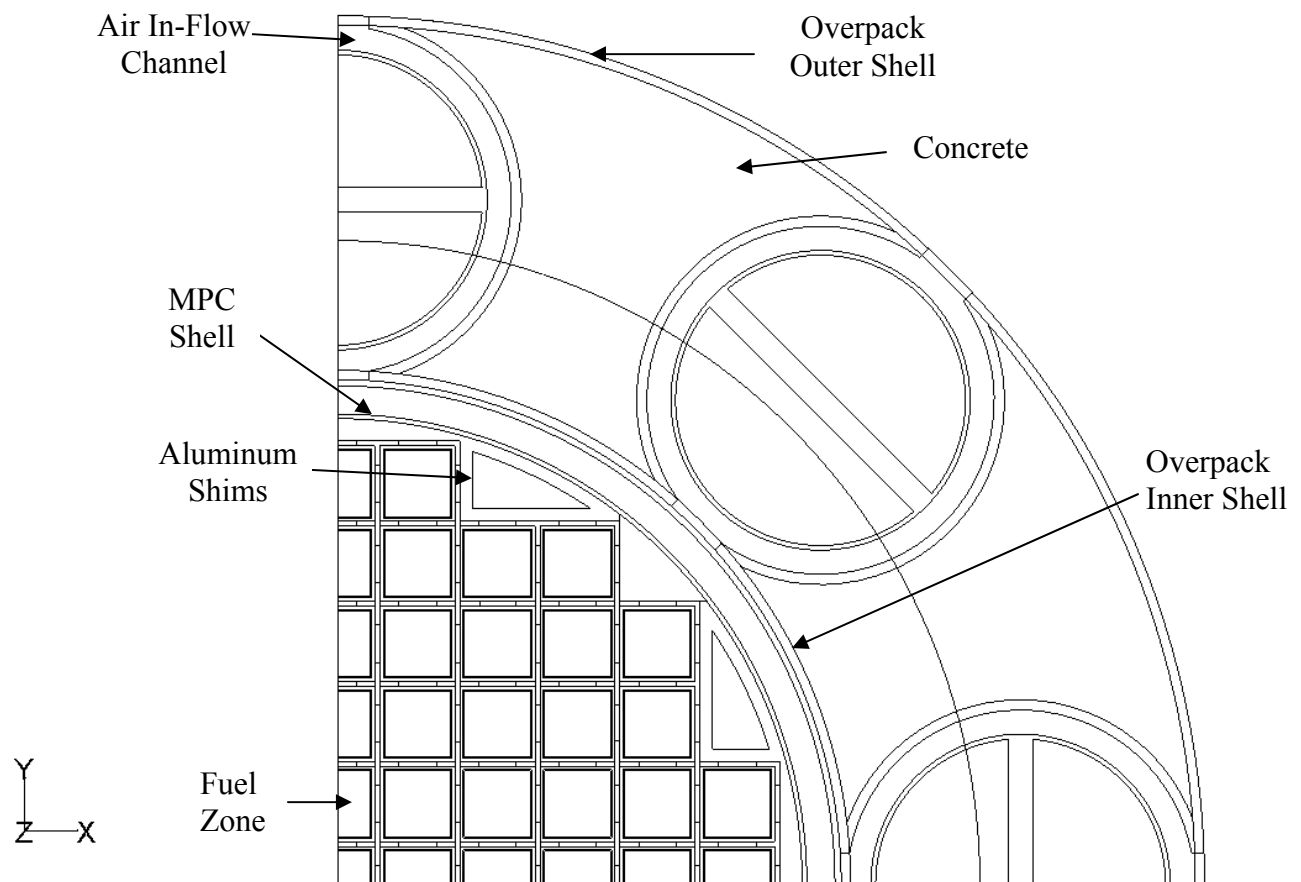


Figure 4.4.2: Planar View of HI-STORM FW MPC-89 Quarter Symmetric 3-D Model

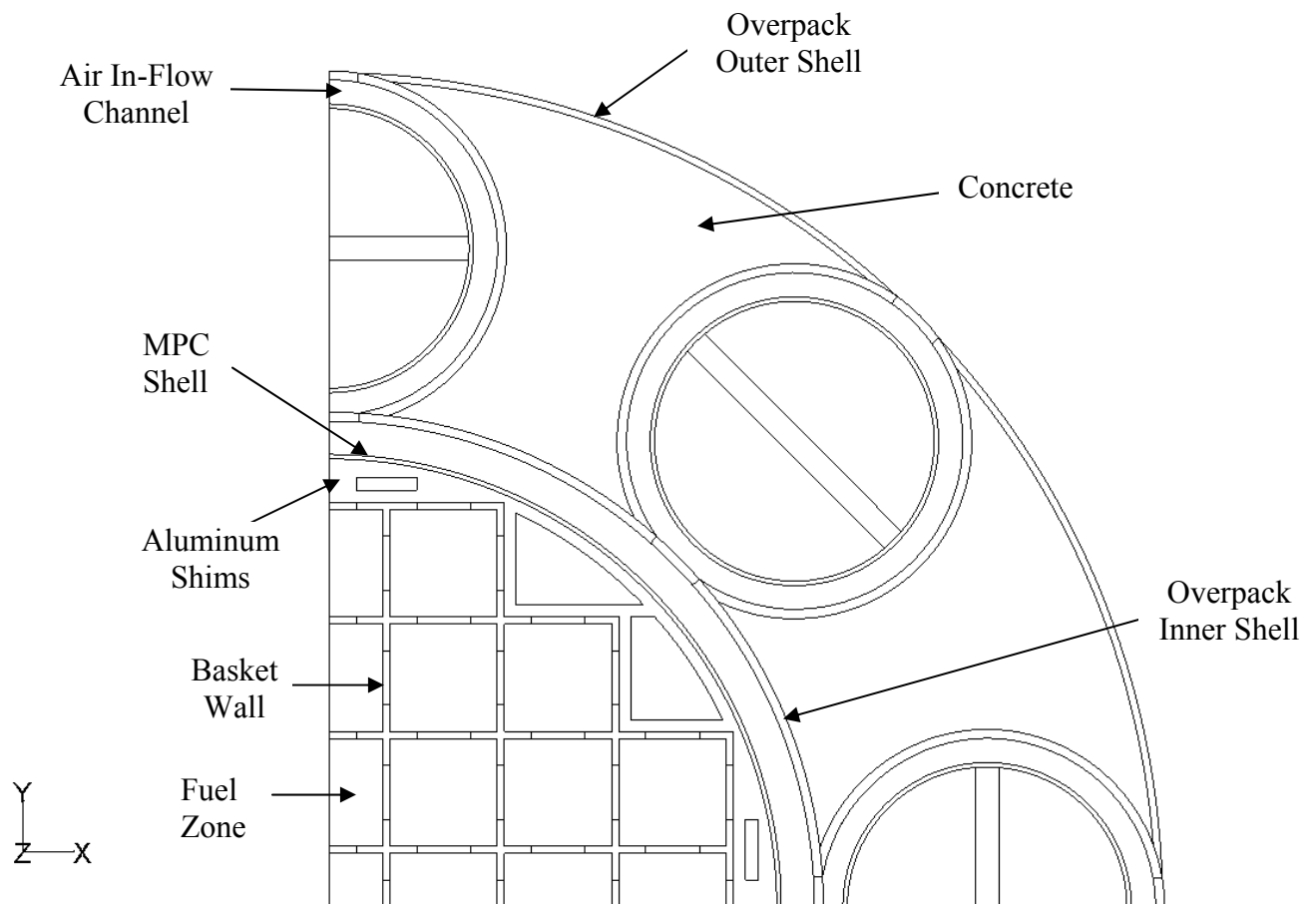


Figure 4.4.3: Planar View of HI-STORM FW MPC-37 Quarter Symmetric 3-D Model

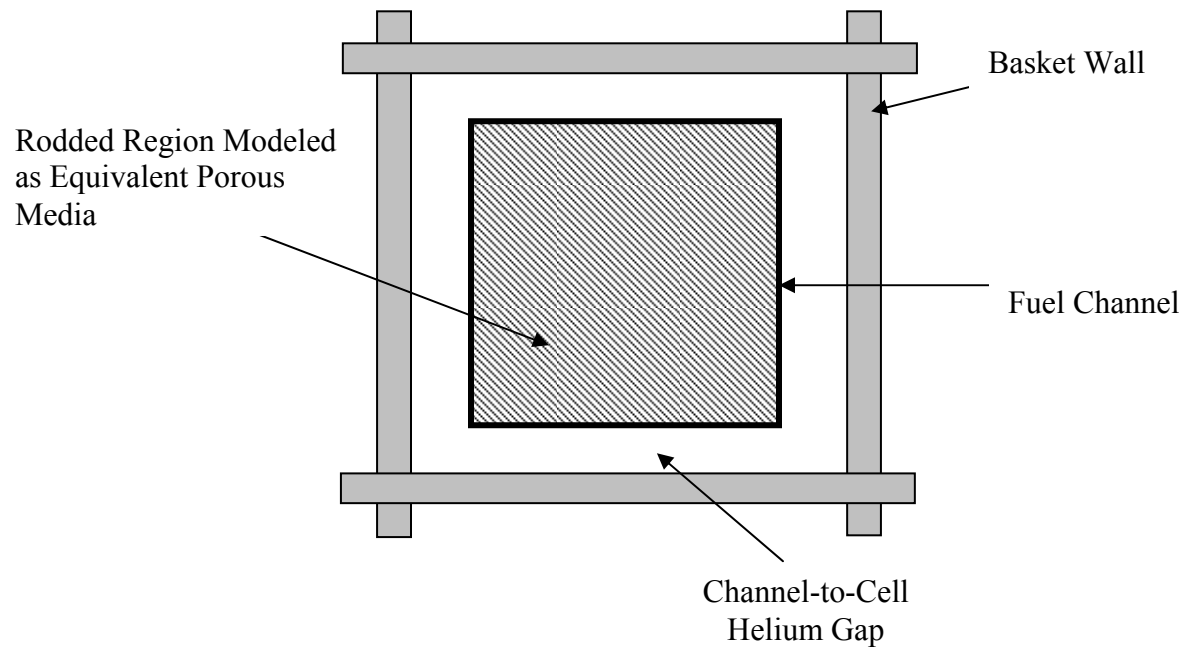
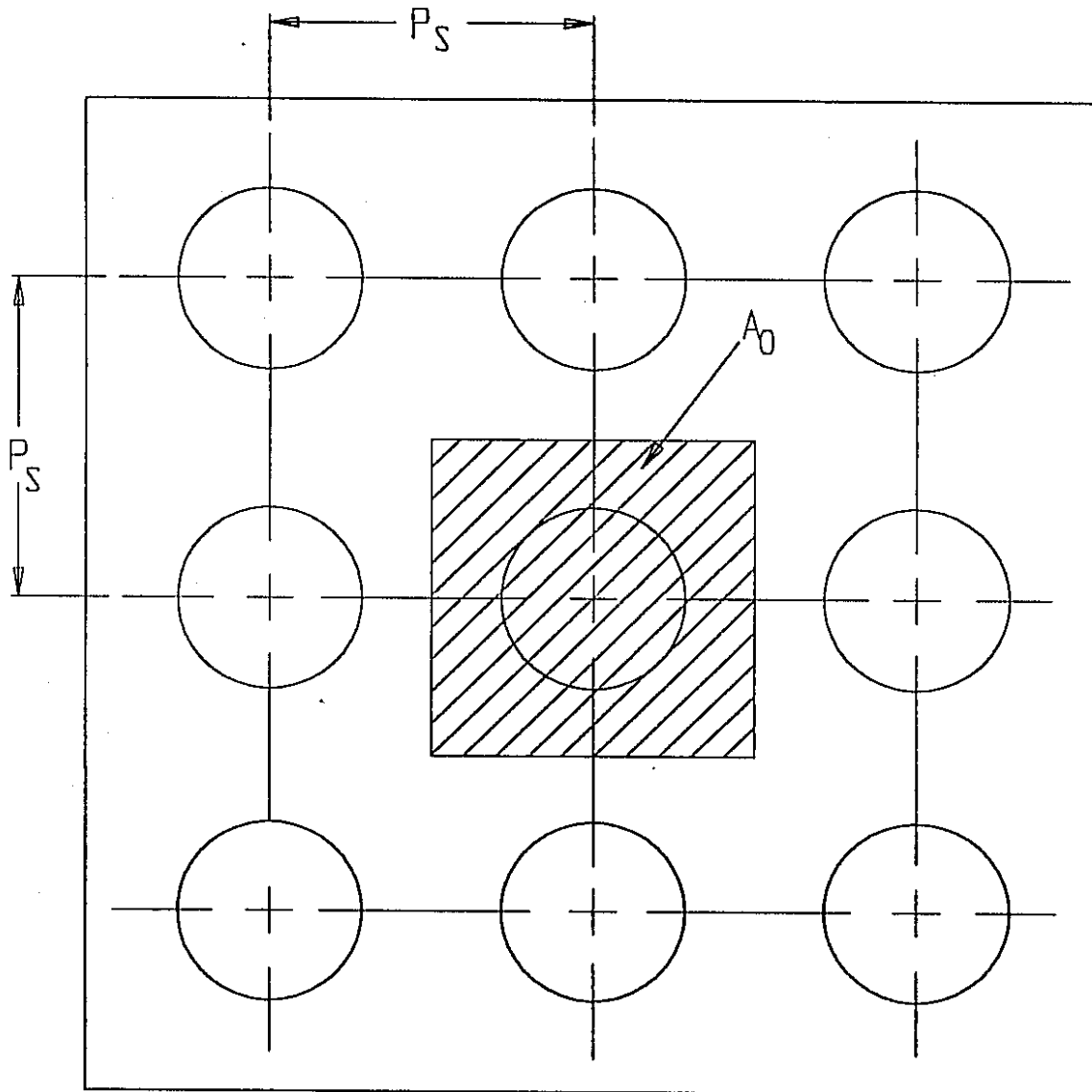


Figure 4.4.4: Closeup View of the MPC-89 Channeled Fuel Spaces



Legend:

$P_s$ : Cask pitch

$A_0$ : Tributary area

Figure 4.4.5: Illustration of a Centrally Located Cask in a Cask Array

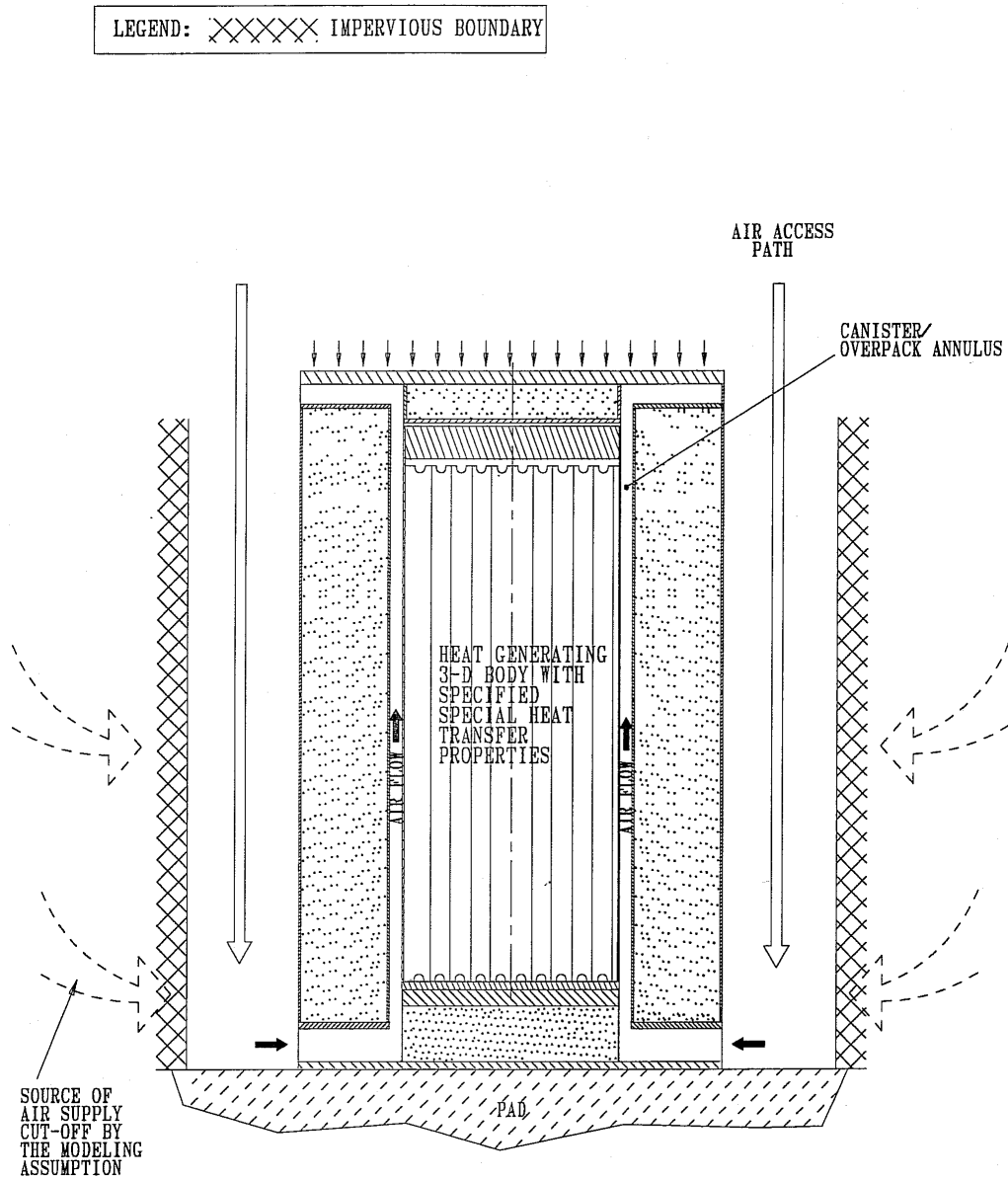


Figure 4.4.6: Illustration of the Hypothetical Square Cavity Thermal Model



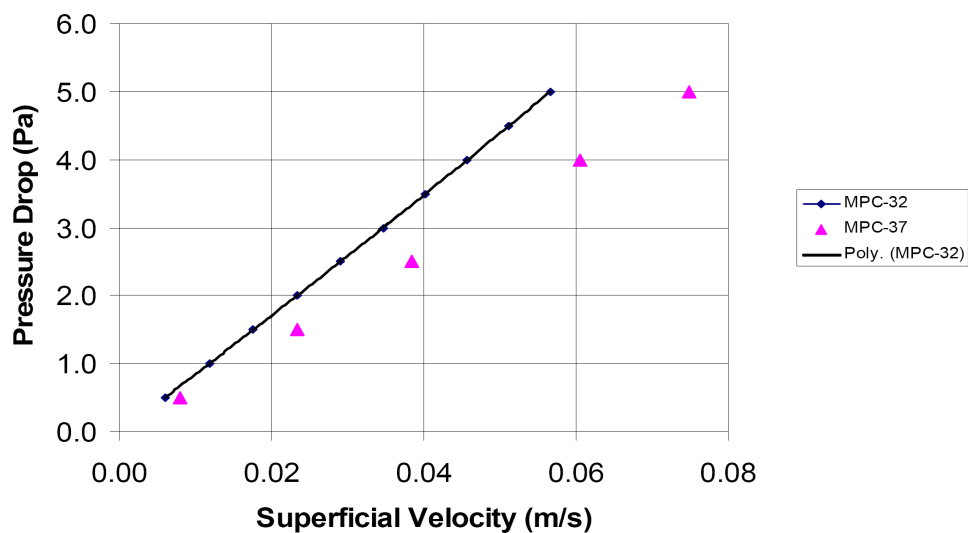


Figure 4.4.7: Storage Cell Pressure Drop as a Function of In-Cell Helium Velocity

## 4.5 THERMAL EVALUATION OF SHORT-TERM OPERATIONS

### 4.5.1 Thermally Limiting Evolutions During Short-Term Operations

Prior to placement in a HI-STORM FW overpack, an MPC must be loaded with fuel, outfitted with closures, dewatered, dried, backfilled with helium and transported to the HI-STORM FW module. In the unlikely event that the fuel needs to be returned to the spent fuel pool, these steps must be performed in reverse. Finally, if required, transfer of a loaded MPC between HI-STORM FW overpacks or between a HI-STAR transport overpack and a HI-STORM FW storage overpack must be carried out in a safe manner. All of the above operations, henceforth referred to as “short-term operations”, are short duration events that would likely occur no more than once or twice for an individual MPC.

Chapter 9 provides a description of the typical loading steps involved in moving nuclear fuel from the spent fuel pool to dry storage in the HI-STORM FW system. The transition from a wet to a dry environment, to comply with ISG-11, Rev. 3, must occur without exceeding the short-term operation temperature limits (see Table 4.3.1).

The loading steps that present the limiting thermal condition during short term operations for the fuel are those when either one or both of the following conditions exist:

- i. The MPC’s fuel storage space is evacuated of fluids resulting in a significant decrease in internal heat transmission rates. This condition obtains if the vacuum drying method for removing moisture from the canister is employed.
- ii. The removal of heat from the external surfaces of the MPC is impeded because of the air gap between the canister and HI-TRAC VW. This condition exists, for example, when the loaded MPC is being moved inside HI-TRAC VW for staging and transfer of the MPC to the HI-STORM FW overpack.

In this section, the thermally limiting scenarios during short-term operations are identified and analyzed.

Because onsite transport of the MPC occurs with the HI-TRAC VW in the vertical orientation, the thermosiphon action within the MPC is preserved at all times. The only (rare) departure from a purely vertical orientation occurs if a tilting of the HI-TRAC VW is needed to clear an obstruction such as a low egress bay door opening at a plant. In such a case the operational imperative for HI-TRAC VW tilting must be ascertained and the permissible duration of non-vertical configuration must be established on a site-specific basis and compliance with the thermal limits of ISG-11 [4.1.4] must be demonstrated as a part of the site-specific safety evaluation under 10CFR72.212.

## 4.5.2 HI-TRAC VW Thermal Model

### 4.5.2.1 On-Site Transfer

The HI-TRAC VW transfer cask is used to load and unload the HI-STORM FW concrete storage overpack, including onsite transport of the MPCs from the loading facility to an ISFSI pad. Within a loaded HI-TRAC VW, heat generated in the MPC is transported from the contained fuel assemblies to the MPC shell through the fuel basket and the basket-to-shell gaps via conduction and thermal radiation. From the outer surface of the MPC to the ambient atmosphere, heat is transported within across multiple concentric layers, representing the air gap, the HI-TRAC VW inner shell, the lead shielding, the HI-TRAC VW outer shell, the water jacket space and the jacket shell. From the surface of the HI-TRAC VW's enclosure shell heat is rejected to the atmosphere by natural convection and radiation.

A small diametral gap exists between the outer surface of the MPC and the inner surface of the HI-TRAC VW overpack which may be filled with water during an operational state to serve as a heat sink and radiation absorber. The water jacket, which provides neutron shielding for the HI-TRAC VW overpack, surrounds the outer cylindrical steel wall of the HI-TRAC VW body. Heat is transported through the water jacket by a combination of conduction through steel ribs and convection heat transfer in the water spaces. The bottom face of the HI-TRAC VW is in contact with a supporting surface which is a thermal heat sink. This face is conservatively modeled as an insulated surface. The HI-TRAC VW is an open top construction which is modeled as an opening to allow air exchange with the ambient.

The HI-TRAC VW Transfer Cask thermal analysis is based on a detailed heat transfer model that conservatively accounts for all modes of heat transfer in the MPC and HI-TRAC VW. The thermal model incorporates several conservative features listed below:

- i. Severe levels of environmental factors - bounding ambient temperature, 32.2°C (90°F), and constant solar flux - were coincidentally imposed on the thermal design. A bounding solar absorptivity of 0.85 is applied to all exposed surfaces.
- ii. The HI-TRAC VW Transfer Cask-to-MPC annular gap is analyzed based on the nominal design dimensions. No credit is considered for the gap reduction that would occur as a result of differential thermal expansion with design basis fuel at hot conditions. The MPC is considered to be concentrically aligned with the cask cavity and the annulus is filled with air. This scenario maximizes thermal resistance.
- iii. The HI-TRAC VW baseplate is in thermally communicative contact with supporting surfaces. For conservatism an insulated boundary condition is applied to the baseplate.
- iv. The HI-TRAC VW fluid columns (namely air in the annulus and water in the water jacket) are allowed to move. In other words natural convection heat transfer by annulus air and water is credited in the analysis.

- v. To maximize lateral resistance to heat dissipation in the fuel basket conservatively postulated 0.8 mm full length panel gaps are assumed at all intersections. This approach is similar to the approach in the approved HI-STAR 180 Package in Docket 71-9325. The shims installed in the MPC peripheral spaces (See MPC-37 and MPC-89 drawings in Section 1.5) are explicitly modeled. For conservatism reasonably bounding gaps (2.5 mm basket-to-shims and 2.5 mm shims-to-shell) are incorporated in the thermal models.
- vi. The Raleigh number of air flow in the annulus between the MPC and HI-TRAC VW indicates that the flow regime in this region is laminar. Therefore, the air flow in this region is modeled as laminar in the thermal model.

The grid deployed in the HI-TRAC VW thermal model is confirmed to be grid independent through mesh sensitivity studies. The studies refined the radial mesh in HI-TRAC VW annulus and water jacket regions. The thermal solutions obtained show that the temperatures are essentially unchanged.

To evaluate on-site transfer operations in a conservative manner a HI-TRAC VW thermal model is constructed under the limiting scenario of fuel storage in the minimum height MPC-37 (See Section 4.4.1.5) and limiting Pattern A heat load specified in Chapter 1, Section 1.2 (See Section 4.4.4). The model adopts the MPC thermal modeling methodology described in Section 4.4 and the properties of design basis 14x14 Ft. Calhoun fuel defined in Table 4.4.1 under the limiting fuel storage scenario cited above. Results of on-site transfer analyses are provided in Subsection 4.5.4.3.

#### **4.5.2.2 Grid Sensitivity Studies**

Cognizant to the grid sensitivity studies performed for the HI-STORM FW System discussed in Section 4.4, a similar study is performed for the HI-TRAC VW System. This study is also performed in accordance with the ASME V&V method [4.4.3]. The grid sensitivity study is performed for the limiting thermal scenario i.e. MPC-37 with minimum fuel length and loaded with pattern A. All the three meshes used for this study satisfy the recommended criterion of 1.3 as the grid refinement factor [4.4.3]. The predicted PCT from these three meshes is essentially the same and are reported in the table below:

<b>Mesh No</b>	<b>Total Mesh Size</b>	<b>PCT (°C)</b>	<b>Permissible Limit (°C)</b>	<b>Clad Temperature Margin (°C)</b>
1 (Licensing Basis Mesh)	1,267,474	389	400	11
2	2,678,012	390	400	10
3	5,797,030	389	400	11

The solutions from these grids are in the asymptotic range. The finest mesh (Mesh 3) has about 4.6 times the total mesh size of the licensing basis mesh (Mesh 1). Even with such a large mesh refinement, the PCT is essentially same for all the three meshes. Since the difference of PCT for all these meshes is close to zero, it indicates that an oscillatory convergence or that the “exact” solution has been attained [4.5.1]. To provide further assurance of convergence, grid convergence index (GCI), which is a measure of the solution uncertainty, is computed as 0.566%. The apparent order of the method is calculated as 1.2.

Based on the above results, it can be concluded that the Mesh 1 is reasonably converged and is adopted as the licensing basis converged mesh.

#### **4.5.2.3 Vacuum Drying**

The initial loading of SNF in the MPC requires that the water within the MPC be drained and replaced with helium. For MPCs containing moderate burnup fuel assemblies only, this operation may be carried out using the conventional vacuum drying approach upto design basis heat load. In this method, removal of moisture from the MPC cavity is accomplished by evacuating the MPC after completion of MPC draining operation. Vacuum drying of MPCs containing high burnup fuel assemblies is permitted up to threshold heat loads defined in Table 4.5.1. High burnup fuel drying in MPCs generating greater than threshold heat load is performed by a forced flow helium drying process as discussed in Section 4.5.4.

Prior to the start of the MPC draining operation, both the HI-TRAC VW 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. To minimize fuel temperatures during vacuum drying operations the HI-TRAC VW annulus must be water filled. The necessary operational steps required to ensure this requirement are set forth in Chapter 9.

A 3-D FLUENT thermal model of the MPC is constructed in the same manner as described in

Section 4.41. The principal input to this model is the effective conductivity of fuel under vacuum drying operations. To bound the vacuum drying operations the effective conductivity of fuel is computed assuming the MPC is filled with water vapor at a very low pressure (1 torr). The methodology for computing the effective conductivity is given in Section 4.4.1 and effective properties of design basis fuel under vacuum conditions tabulated in Table 4.5.8. To ensure a conservative evaluation the thermal model is incorporated with the following assumptions:

- i. Bounding steady-state condition is reached with the MPC decay heat load set equal to the limiting heat load (Pattern A in Table 1.2.3 and 1.2.4) for MPCs fueled with Moderate Burnup Fuel and threshold heat load defined in Table 4.5.1 for MPCs fueled with one or more High Burnup fuel assemblies.
- ii. The external surface of the MPC shell is postulated to be at the boiling temperature of water 100°C (212°F).
- iii. The bottom surface of the MPC is insulated.
- iv. MPC internal convection heat transfer is suppressed.

Results of vacuum condition analyses are provided in Subsection 4.5.4.1.

#### **4.5.3 Maximum Time Limit During Wet Transfer Operations**

In accordance with NUREG-1536, water inside the MPC cavity during wet transfer operations is not permitted to boil. This requirement is met by imposing time limits for fuel to remain submerged in water after a loaded HI-TRAC VW cask is removed from the pool.

Fuel loading operations are typically conducted with the HI-TRAC VW and its contents (water filled MPC) submerged in pool water. Under these conditions, the HI-TRAC VW is essentially at the pool water temperature. When the HI-TRAC VW transfer cask and the loaded MPC under water-flooded conditions is removed from the pool, the water, fuel, MPC and HI-TRAC VW metal absorb the decay heat emitted by the fuel assemblies. This results in a slow temperature rise of the HI-TRAC VW with time, starting from an initial (pool water) temperature. The rate of temperature rise is limited by the thermal inertia of the HI-TRAC VW system. To enable a bounding heat-up rate determination, the following conservative assumptions are utilized:

- i. Heat loss by natural convection and radiation from the exposed HI-TRAC VW surfaces to ambient air is neglected (i.e., an adiabatic heat-up calculation is performed).
- ii. Design maximum heat input from the loaded fuel assemblies is assumed.

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<sup>1</sup> The MPC thermal model adopted for vacuum drying analysis in this sub-section includes the gap between the intersecting basket panels as 0.4 mm. A sensitivity study of the most limiting thermal scenario (least margins to fuel temperature limit) of vacuum drying condition is performed with this gap as 0.8 mm and discussed in Sub-section 4.5.4.4.

- iii. The shortest allowable HI-TRAC VW is credited in the analysis to impart the lowest thermal inertia on the system, which will result in the highest rate of temperature rise.
- iv. The water mass in the MPC cavity is understated.

Table 4.5.3 summarizes the weights and thermal inertias of several components in the loaded HI-TRAC VW transfer cask that corresponds to the shortest allowable fuel assembly. The rate of temperature rise of the HI-TRAC VW transfer cask and contents during an adiabatic heat-up is governed by the following equation:

$$\frac{dT}{dt} = \frac{Q}{C_h}$$

where:

- Q = conservatively bounding heat load (Btu/hr)
- C<sub>h</sub> = thermal inertia of a loaded HI-TRAC VW (Btu/°F)
- T = temperature of the HI-TRAC VW transfer cask (°F)
- t = time after MPC lid is placed (hr)

From this adiabatic rate of temperature rise estimate, the maximum allowable time duration (t<sub>max</sub>) for fuel to be submerged in water is determined as follows:

$$t_{\max} = \frac{T_{\text{boil}} - T_{\text{initial}}}{(dT/dt)}$$

where:

- T<sub>boil</sub> = boiling temperature of water (equal to 212°F at the water surface in the MPC cavity)
- T<sub>initial</sub> = initial pool water temperature when the lid is placed on the MPC (°F)

The time-to-boil clock starts when the lid is placed on the MPC and the HI-TRAC is in the spent fuel pool, while it ends when the MPC is drained (See section 9.2.4). Table 4.5.4 provides a summary of t<sub>max</sub> at several representative initial temperatures. The time-to-boil calculations are conservatively performed for the HI-TRAC VW that corresponds to shortest allowed fuel assembly since lowerbound thermal inertia results in lower time limits. A site-specific time-to-boil calculation can be performed using the above equations based on the actual canister heat load and thermal inertia of the specific HI-TRAC VW System.

As set forth in the HI-STORM FW operating procedures, in the unlikely event that the maximum allowable time provided in Table 4.5.3 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 MPC lid ports 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-°F)

$T_{\max}$  = suitably limiting temperature below boiling (°F)

$T_{in}$  = water supply temperature to MPC

#### 4.5.4 Analysis of Limiting Thermal States During Short-Term Operations

##### 4.5.4.1 Vacuum Drying

The vacuum drying option is evaluated for the two limiting scenarios defined in Section 4.5.2.2 to address Moderate Burnup Fuel under limiting heat load (Pattern A) and High Burnup Fuel under threshold heat load defined in Table 4.5.1. The principle objective of the analysis is to ensure compliance with ISG-11 temperature limits. For this purpose 3-D FLUENT thermal models of the MPC-37 and MPC-89 canisters are constructed as described in Section 4.5.2.2 and bounding steady state temperatures computed. The results are tabulated in Tables 4.5.6 and 4.5.7. The results show that the cladding temperatures comply with the ISG-11 limits for moderate and high burnup fuel in Table 4.3.1 by robust margins.

##### 4.5.4.2 Forced Helium Dehydration

To reduce moisture to trace levels in the MPC using a Forced Helium Dehydration (FHD) system, a conventional, closed loop dehumidification system consisting of a condenser, a demister, 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 turbulence. Demisterization to the 3 torr vapor pressure criteria required by NUREG 1536 is assured by verifying that the helium temperature exiting the demister is maintained at or below the psychrometric threshold of 21°F for a minimum of 30 minutes. Appendix 2.B of [4.1.8] provides a detailed discussion of the design criteria and operation of 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 in Table 2.2.3. 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 long term 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 onsite transfer. As a result, if



the FHD machine fails then the peak fuel cladding temperatures will approximate the value reached during normal onsite transfer, discussed below.

#### 4.5.4.3 Normal On-site Transfer

An MPC-37 situated inside a HI-TRAC VW is evaluated under the design heat load defined in Section 1.2. The MPC-37 is evaluated because it yields the highest fuel and cask temperatures (See Table 4.4.2). This scenario is analyzed using the same 3D FLUENT model of the MPC-37 articulated in Section 4.4 for normal storage with due recognition of it situated in the HI-TRAC VW transfer cask. The HI-TRAC VW model discussed in Section 4.5.2 is adopted to construct a global model of an MPC-37 situated inside the HI-TRAC VW and dissipating heat by natural convection and radiation to ambient air.

While the duration of onsite transport is generally short to preclude the MPC and HI-TRAC VW from reaching a steady-state, a conservative approach is adopted herein by assuming steady state maximum temperatures are reached. The principle objectives of the HI-TRAC VW analyses are to demonstrate:

- i) Cladding integrity
- ii) Confinement integrity
- iii) Neutron shield integrity

The appropriate criteria are provided in Tables 2.2.1 (pressure limits) and 2.2.3 (temperature limits).

The results of thermal analyses tabulated in Table 4.5.2 show that the cladding temperatures are below the ISG-11 temperature limits of High and Moderate Burnup Fuel (Table 4.3.1). Actual margins during HI-TRAC VW operations will be much larger due to the many conservative assumptions incorporated in the analysis.

The water in the water jacket surrounding the HI-TRAC VW body provides necessary neutron shielding. During normal handling and onsite transfer operations this shielding water is contained within the water jacket at elevated internal pressure. The water jacket is equipped with two pressure relief devices set to an adequately high pressure to prevent boiling. Under HI-TRAC VW operations, the bulk temperature of water remains below the temperature limit specified in Table 2.2.3. Accordingly, water is in the liquid state and the neutron shielding function is maintained. The cladding, neutron shield and HI-TRAC VW component temperatures are provided in Table 4.5.2. The confinement boundary integrity is evaluated in the Section 4.5.6.

#### 4.5.4.4 Effect of Increase in Basket Panel Gap

As described in Subsection 4.5.2.3, a sensitivity study is performed for the vacuum drying

condition of high burnup fuel at threshold heat load with the basket panel notch gap equal to 0.8 mm. The results of the steady state analysis vacuum drying condition are summarized in Table 4.5.10. The PCT and cask component temperatures during vacuum drying are below their respective temperature limits. Therefore, the effect of increasing the panel gap is small and leaves sufficient safety margins during vacuum drying conditions.

#### 4.5.4.5 Evaluation of 15x15I Short Fuel Assembly

##### (a) On-Site Transfer

The thermal evaluations described in this sub-section are performed for the 15x15I short fuel height of the HI-TRAC VW, similar to the evaluations performed for HI-STORM FW System in Sub-Section 4.4.1.7. The thermal model is exactly the same as that described in sub-section 4.5.2 with the following exceptions:

1. The fuel basket height is conservatively modeled lower i.e. equal to 149”.
2. The flow resistance through the fuel assemblies is based on the calculations in sub-section 4.4.1.7.
3. The effective fuel thermal properties specific to 15x15I fuel assembly are used (see Table 4.4.1)

The converged mesh from Sub-section 4.5.2.2 is utilized to evaluate the normal on-site transfer of 15x15I short MPC-37 with bounding heat load pattern A in HI-TRAC VW. The results from the steady state analysis are tabulated in Table 4.5.11. The fuel temperatures are bounded by the licensing basis results reported in Table 4.5.2 for the minimum MPC-37 (short). The temperatures of other cask components are similar to the licensing basis evaluation. Therefore, the minimum MPC-37 (short) is adopted for all the licensing evaluations.

##### (b) Vacuum Drying

The surface area of the MPC for heat transfer reduces for a shorter MPC height. Therefore, the 15x15I short fuel assembly under the “minimum” category defined in Table 2.1.1 warrants a thermal analysis to show compliance with regulatory limits. A steady state vacuum drying thermal evaluation is performed at threshold heat load for the 15x15I short MPC based on the same methodology described in Sub-section 4.5.2.3. The results of this analysis are reported in Table 4.5.12. The results show that the cladding temperatures comply with the ISG-11 limits for moderate and high burnup fuel in Table 4.3.1 by robust margins.

#### 4.5.4.6 Thermal Evaluation of Various Extruded Basket Shim Design Options

To allow flexibility in fabrication, the licensing drawings provide various options to install extruded basket shims between the basket and inside of the MPC wall. A summary of various allowable options is provided in Table 1.2.9. From a thermal standpoint, emissivity of shims and the average total gap between the basket and extruded shim and the extruded shim and MPC shell are critical to heat transfer from the fuel basket to MPC. As noted in Table 1.2.9, solid shim aluminum plates may be placed between the basket and extruded shim to ensure the criteria on

the gap in the basket periphery is met.

A series of thermal analyses to address all the allowable design options presented in Table 1.2.9 are performed and discussed below:

(a) On-Site Transfer

All the design options presented in Table 1.2.9 are evaluated below:

- i) Option 1: This option corresponds to the licensing basis analysis presented in Sub-section 4.5.4.3. The predicted temperature and MPC cavity pressure using this option are reported in Tables 4.5.2 and 4.5.5 respectively. All the licensing basis thermal evaluations for HI-TRAC documented in this chapter are performed based on Option 1 in Table 1.2.9 since it results in the most limiting PCT and MPC pressure.
- ii) Option 2: The basket extruded shims are fabricated to provide a loose fit in the basket peripheral space. If the average radial total cold gap in the basket periphery exceeds the gap in Option 1 (Table 1.2.9), solid aluminum shim plates are placed in the space between the basket and extruded shims. The average radial total cold gap after the placement of solid shim plates must be restricted to that provided for Option 2 in Table 1.2.9. The gap between the basket and extruded shims with a solid shim plate is modeled as resistance element with an effective thermal conductivity. A two-dimensional evaluation is performed to determine this effective thermal conductivity and documented in the calculation package [4.1.9].

The calculated effective thermal conductivity in the basket periphery region from the two-dimensional model is used as an input to the licensing basis three-dimensional thermal model described above. All other input parameters remain unchanged. The predicted PCT for this option under normal onsite transfer in HI-TRAC is tabulated in Table 4.5.13.

- iii) Option 3: The effective thermal conductivity in the space between the basket and extruded shims is bounded by that corresponding to Option 4 [4.1.9]. This design option is bounded by Option 4.
- iv) Option 4: As discussed earlier, emissivity of extruded shims have an impact on the radiation heat transfer within the MPC and therefore must be evaluated. This option corresponds to low emissive extruded shims. If the average radial total cold gap in the basket periphery exceeds the gap in Option 3 (Table 1.2.9), solid aluminum shim plates are placed in the space between the basket and extruded shims. The average radial total cold gap after the placement of solid shim plates must be restricted to that provided for Option 4 in Table 1.2.9. The gap between the basket and extruded shims with a solid shim plate is modeled as resistance element with an effective thermal conductivity. A two-dimensional evaluation is

performed to determine this effective thermal conductivity and documented in the calculation package [4.1.9].

The calculated effective thermal conductivity in the basket periphery region from the two-dimensional model is used as an input to the licensing basis three-dimensional thermal model described above. All other input parameters remain unchanged. The predicted PCT for this option under normal onsite transfer in HI-TRAC is tabulated in Table 4.5.13.

- v) Option 5: The effective thermal conductivity in the space between the basket and extruded shims is bounded by that corresponding to Option 6 [4.1.9]. This design option is bounded by Option 6.
- vi) Option 6: As discussed earlier, emissivity of extruded shims have an impact on the radiation heat transfer within the MPC and therefore must be evaluated. If the average radial total cold gap in the basket periphery exceeds the gap in Option 5 (Table 1.2.9), solid aluminum shim plates are placed in the space between the basket and extruded shims. The average radial total cold gap after the placement of solid shim plates must be restricted to that provided for Option 6 in Table 1.2.9. The gap between the basket and extruded shims with a solid shim plate is modeled as resistance element with an effective thermal conductivity. A two-dimensional evaluation is performed to determine this effective thermal conductivity and documented in the calculation package [4.1.9].

The calculated effective thermal conductivity in the basket periphery region from the two-dimensional model is used as an input to the licensing basis three-dimensional thermal model described above. All other input parameters remain unchanged. The predicted PCT for this option under normal onsite transfer in HI-TRAC is tabulated in Table 4.5.13.

The PCTs from the above sensitivity studies tabulated in Table 4.5.13 demonstrate that Option 1 is thermally the most limiting design option. Therefore, it is adopted for all the normal, off-normal and accident condition thermal evaluations. Since MPC-37 is the limiting thermal basket, these design options can be extended to MPC-89 fuel basket.

#### (b) Vacuum Drying

A series of thermal analyses to address all the allowable design options presented in Table 1.2.9 are also performed for the most limiting vacuum drying scenario i.e. under threshold heat load. These thermal evaluations are similar to those performed for normal on-site transfer and discussed above.

The PCTs from the sensitivity studies tabulated in Table 4.5.14 demonstrate that Option 1 is thermally the most limiting design option under vacuum drying condition.

#### **4.5.4.7 Sensitivity Study of Basket Extruded Shim Thickness**

All the licensing basis thermal evaluations in this chapter are performed with an inch thick extruded basket shims. A sensitivity study is performed to evaluate the effect of reducing the thickness of basket shims on the fuel, MPC and HI-TRAC component temperature field during normal onsite transfer. The converged mesh from Sub-section 4.5.2.2 is adopted for this study. The thermal model is exactly the same as that discussed in Sub-section 4.5.2 except that the thickness of extruded shims is reduced to 0.5 inches. The results of the sensitivity study are documented in Table 4.5.15. The results demonstrate that the effect of reducing the thickness of extruded shims is small and leaves sufficient margins to temperature limits.

#### **4.5.4.8 Normal On-site Transfer inside a Building**

Normal on-site transfer using the HI-TRAC VW can be carried out inside a building. When HI-TRAC VW is located inside a building, the ambient air temperature inside the building could be higher than the outdoor environment temperature used in the thermal evaluations performed in Subsection 4.5.4.3. To evaluate this scenario, an ambient temperature that corresponds to the maximum indoor air temperature specified in Table 2.2.2 for short term operations is assumed. Since the cask is inside a building, no solar insolation is applied to the cask. A steady state analysis is performed for the limiting thermal scenario of MPC-37 inside the HI-TRAC VW under heat load pattern A. The peak cladding, MPC and the HI-TRAC component temperatures are presented in Table 4.5.9 in addition to the MPC cask cavity pressure. The predicted component temperatures and MPC cavity pressure are below their respective temperatures and pressure for outdoor environment presented in Tables 4.5.2 and 4.5.5 respectively. Therefore, the normal on-site transfer of a HI-TRAC outside the building and with solar insolation as evaluated in Subsection 4.5.4.3 is the limiting thermal condition.

#### **4.5.5 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 under a reflood rate of 3715 lb/hr. Limiting the water

reflood rate to this amount or less would prevent exceeding the MPC design pressure.

#### **4.5.6 Maximum Internal Pressure (Load Case NB in Table 2.2.7)**

After fuel loading and vacuum drying, but prior to installing the MPC closure ring, the MPC is initially filled with helium. During handling and on-site transfer operations in the HI-TRAC VW transfer cask, the gas temperature will correspond to the thermal conditions within the MPC analyzed in Section 4.5.4.3. Based on this analysis the MPC internal pressure is computed under the assumption of maximum helium backfill specified in Table 4.4.8 and confirmed to comply with the short term operations pressure limit in Table 2.2.1. The results are tabulated in Table 4.5.5.

TABLE 4.5.1 THRESHOLD HEAT LOADS UNDER VACUUM DRYING OF HIGH BURNUP FUEL (See Figures 1.2.1 and 1.2.2)			
MPC-37			
Number of Regions: 3			
Number of Storage Cells: 37			
Maximum Heat Load: 34.36			
Region No.	Decay Heat Limit per Cell, kW	Number of Cells per Region	Decay Heat Limit per Region, kW
1	0.80	9	7.2
2	0.97	12	11.64
3	0.97	16	15.52
MPC-89			
Number of Regions: 3			
Number of Storage Cells: 89			
Maximum Heat Load: 34.75			
Region No.	Decay Heat Limit per Cell, kW	Number of Cells per Region	Decay Heat Limit per Region, kW
1	0.35	9	3.15
2	0.35	40	14.00
3	0.44	40	17.60

Table 4.5.2

## HI-TRAC VW TRANSFER CASK STEADY STATE MAXIMUM TEMPERATURES

<b>Component</b>	<b>Temperature, °C (°F)</b>
Fuel Cladding	389 (732)
MPC Basket	374 (705)
Basket Periphery	299 (570)
Aluminum Basket Shims	272 (522)
MPC Shell	247 (477)
MPC Lid <sup>Note 1</sup>	240 (464)
HI-TRAC VW Inner Shell	138 (280)
HI-TRAC VW Radial Lead Gamma Shield	138 (280)
Water Jacket Bulk Water	129 (264)

Note 1: Maximum section average temperature is reported.



Table 4.5.3			
HI-TRAC VW TRANSFER CASK LOWERBOUND WEIGHTS AND THERMAL INERTIAS <sup>Note 1</sup>			
Component	Weight (lbs)	Heat Capacity (Btu/lb-°F)	Thermal Inertia (Btu/°F)
Lead	45627	0.031	1414
Carbon Steel	43270	0.1	4327
Stainless Steel	19561	0.12	2347
Aluminum	6734	0.207	1394
Metamic-HT	7349	0.22	1617
Fuel	46250	0.056	2590
MPC Cavity Water	6611	0.999	6604
Total	175402	-	20294
Note 1: Values presented in this table are based on the short HI-TRAC VW height determined in accordance with Table 3.2.1 using a PWR fuel height of 157”.			

Table 4.5.4	
MAXIMUM ALLOWABLE TIME FOR WET TRANSFER OPERATIONS <sup>Note 1</sup>	
Initial temperature °F	Time Duration (hr)
100	14.2
110	12.9
120	11.6
130	10.4
140	9.1
150	7.8
Note 1: The time-to-boil limits provided herein are based on the HI-TRAC VW for a PWR fuel height of 157” and maximum design basis heat load. A site-specific calculation based on the methodology described in Section 4.5.3 can be performed to determine the time-to-boil limits.	

Table 4.5.5	
MPC CONFINEMENT BOUNDARY PRESSURE UNDER ON-SITE TRANSPORT	
Condition	Pressure (psig)
Initial backfill pressure (at 70°F) (Tech. Spec. maximum in Table 4.4.8)	45.5
Maximum pressure	100.7

Table 4.5.6		
MAXIMUM TEMPERATURES OF MPC-37 DURING VACUUM DRYING CONDITIONS		
Component	Temperatures @DB Heat Load <sup>Note 1</sup> °C (°F)	Temperatures @ Threshold Heat Load <sup>Note 2</sup> °C (°F)
Fuel Cladding	480 (896)	384 (723)
MPC Basket	464 (867)	367 (693)
Basket Periphery	357 (675)	288 (550)
Aluminum Basket Shims	278 (532)	232 (450)
MPC Shell	156 (313)	142 (288)
MPC Lid <sup>Note 3</sup>	107 (225)	100 (212)
Note 1: Addresses vacuum drying of Moderate Burnup Fuel under limiting heat load (Pattern A) defined in Section 1.2.		
Note 2: Addresses vacuum drying of High Burnup Fuel under threshold heat load (Table 4.5.1).		
Note 3: Maximum section temperature reported.		

Table 4.5.7

## MAXIMUM TEMPERATURES OF MPC-89 DURING VACUUM DRYING CONDITIONS

<b>Component</b>	<b>Temperatures @DB Heat Load<sup>Note 1</sup></b> <b>°C (°F)</b>	<b>Temperatures @ Threshold Heat Load<sup>Note 2</sup></b> <b>°C (°F)</b>
Fuel Cladding	464 (867)	376 (709)
MPC Basket	449 (840)	359 (678)
Basket Periphery	348 (658)	286 (547)
Aluminum Basket Shims	275 (527)	232 (450)
MPC Shell	158 (316)	144 (291)
MPC Lid <sup>Note 3</sup>	127 (261)	110 (230)

Note 1: Addresses vacuum drying of Moderate Burnup Fuel under Design Basis heat load defined in Section 1.2.

Note 2: Addresses vacuum drying of High Burnup Fuel under threshold heat load (Table 4.5.1).

Note 3: Maximum section temperature reported.

Table 4.5.8

EFFECTIVE CONDUCTIVITY OF DESIGN BASIS FUEL<sup>Note 1</sup> UNDER VACUUM DRYING OPERATIONS  
(Btu/hr-ft-°F)

Temperature (°F)	Planar	Axial
200	0.111	0.737
450	0.273	0.805
700	0.538	0.900
1000	0.977	1.040

Note 1: Ft. Calhoun 14x14 fuel is defined as the design basis fuel under the limiting condition of fuel storage in the minimum height MPC-37 (See Table 4.4.2).

Table 4.5.9	
STEADY STATE MAXIMUM TEMPERATURES AND MPC CAVITY PRESSURE FOR A HI-TRAC VW TRANSFER CASK INSIDE A BUILDING <sup>1</sup>	
Component	Temperature, °C (°F)
Fuel Cladding	387 (729)
MPC Basket	373 (703)
Basket Periphery	297 (567)
Aluminum Basket Shims	271 (520)
MPC Shell	246 (475)
MPC Lid <sup>Note 1</sup>	238 (460)
HI-TRAC VW Inner Shell	135 (275)
HI-TRAC VW Radial Lead Gamma Shield	134 (273)
Water Jacket Bulk Water	126 (259)
MPC Cavity Pressure, kPa (psig)	
No Rod Rupture	691.4 (100.3)
Note 1: Maximum section average temperature is reported.	

<sup>1</sup> This condition corresponds to a HI-TRAC inside a building at an ambient temperature specified in Table 2.2.2 and without solar insolation.

Table 4.5.10	
EFFECT OF INCREASE IN BASKET PANEL GAP ON MAXIMUM TEMPERATURES OF MPC-37 DURING VACUUM DRYING CONDITION AT THRESHOLD HEAT LOAD	
Component	Temperatures <sup>Note 1</sup> , °C (°F)
Fuel Cladding	389 (732)
MPC Basket	373 (703)
Basket Periphery	292 (558)
Aluminum Basket Shims	234 (453)
MPC Shell	142 (288)
Note 1: The predicted temperatures for the increased panel gap are slightly higher than the licensing basis temperatures for threshold heat load reported in Table 4.5.6.	

Table 4.5.11	
HI-TRAC VW TRANSFER CASK STEADY STATE MAXIMUM TEMPERATURES AND MPC CAVITY PRESSURE FOR 15x15I SHORT FUEL ASSEMBLY	
Component	Temperature, °C (°F)
Fuel Cladding	382 (720)
MPC Basket	366 (691)
Basket Periphery	301 (574)
Aluminum Basket Shims	290 (554)
MPC Shell	254 (489)
MPC Lid <sup>Note 1</sup>	250 (482)
HI-TRAC VW Inner Shell	141 (286)
HI-TRAC VW Radial Lead Gamma Shield	140 (284)
Water Jacket Bulk Water	131 (268)
MPC Cavity Pressure (psig)	
No Rod Rupture <sup>Note 2</sup>	100.3
Note 1: Maximum section average temperature is reported.	
Note 2: The MPC cavity pressure is calculated based on the Tech. Spec. maximum value (see Table 4.4.8).	

Table 4.5.12	
MAXIMUM TEMPERATURES OF MPC-37 DURING VACUUM DRYING CONDITIONS AT THRESHOLD HEAT LOAD FOR 15x15I SHORT FUEL ASSEMBLY	
Component	Temperatures °C (°F)
Fuel Cladding	389 (732)
MPC Basket	369 (696)
Basket Periphery	280 (536)
Aluminum Basket Shims	235 (455)
MPC Shell	149 (300)
MPC Lid <sup>Note 1</sup>	95 (203)
Note 1: Maximum section temperature reported.	

TABLE 4.5.13	
PEAK CLADDING TEMPERATURES FOR EXTRUDED BASKET SHIM DESIGN OPTIONS UNDER NORMAL ONSITE TRANSFER USING HI-TRAC VW	
Design Option <sup>Note 1</sup>	PCT (°C)
Option 1 <sup>Note 2</sup>	389 <sup>Note 3</sup>
Option 2	386
Option 3	Note 4
Option 4	381
Option 5	Note 5
Option 6	387
Notes: (1) All the extruded basket shim design options are summarized in Table 1.2.9. (2) Option 1 corresponds to the licensing basis evaluation documented in Table 4.5.2. (3) Option 1 results in the highest peak cladding temperature and is therefore adopted as the licensing basis option for thermal evaluation of long-term storage in HI-STORM and under all off-normal and accident conditions. (4) Option 3 is bounded by Option 4. (5) Option 5 is bounded by Option 6.	



TABLE 4.5.14	
PEAK CLADDING TEMPERATURES FOR EXTRUDED BASKET SHIM DESIGN OPTIONS UNDER VACCUM DRYING CONDITION AT THRESHOLD HEAT LOAD	
Design Option <sup>Note 1</sup>	PCT (°C)
Option 1 <sup>Note 2</sup>	389 <sup>Note 3</sup>
Option 2	376
Option 3	Note 4
Option 4	362
Option 5	Note 5
Option 6	376
Notes: (1) All the extruded basket shim design options are summarized in Table 1.2.9. (2) Option 1 corresponds to the licensing basis evaluation documented in Table 4.5.10. (3) Option 1 results in the highest peak cladding temperature and is therefore adopted as the licensing basis evaluation. (4) Option 3 is bounded by Option 4. (5) Option 5 is bounded by Option 6.	

Table 4.5.15	
SENSITIVITY STUDY OF REDUCED BASKET SHIM THICKNESS ON HI-TRAC VW TRANSFER MAXIMUM TEMPERATURES DURING NORMAL ON-SITE TRANSFER	
Component	Temperature, °C <sup>Note 2</sup>
Fuel Cladding	391
MPC Basket	376
Basket Periphery	304
Aluminum Basket Shims	277
MPC Shell	250
MPC Lid <sup>Note 1</sup>	240
HI-TRAC VW Inner Shell	138
HI-TRAC VW Radial Lead Gamma Shield	137
Water Jacket Bulk Water	129
Note 1: Maximum section average temperature is reported.	
Note 2: The results are essentially the same as the licensing basis thermal evaluation presented in Table 4.5.2.	

## 4.6 OFF-NORMAL AND ACCIDENT EVENTS

In this Section thermal evaluation of HI-STORM FW System under off-normal and accident conditions defined in Sections 4.6.1 and 4.6.2 is provided. To ensure a bounding evaluation the limiting Pattern A thermal loading scenario defined in Section 4.4.4 is adopted in the evaluation.

### 4.6.1 Off-Normal Events

#### 4.6.1.1 Off-Normal Pressure (Load Case NB in Table 2.2.7)

This event is defined as a combination of (a) maximum helium backfill pressure (Table 4.4.8), (b) 10% fuel rods rupture, (c) limiting fuel storage configuration and (d) off-normal ambient temperature. The principal objective of the analysis is to demonstrate that the MPC off-normal design pressure (Table 2.2.1) is not exceeded. The MPC off-normal pressures are reported in Table 4.6.7. The result is below the off-normal design pressure (Table 2.2.1).

#### 4.6.1.2 Off-Normal Environmental Temperature

This event is defined by a time averaged ambient temperature of 100°F for a 3-day period (Table 2.2.2). The results of this event (maximum temperatures and pressures) are provided in Table 4.6.1 and 4.6.7. The results are below the off-normal condition temperature and pressure limits (Tables 2.2.3 and 2.2.1).

#### 4.6.1.3 Partial Blockage of Air Inlets

The HI-STORM FW system is designed with debris screens installed on the inlet and outlet openings. These screens ensure the air passages are protected from entry and blockage by foreign objects. As required by the design criteria presented in Chapter 2, it is postulated that the HI-STORM FW air inlet vents are 50% blocked. The resulting decrease in flow area increases the flow resistance of the inlet ducts. The effect of the increased flow resistance on fuel temperature is analyzed for the normal ambient temperature (Table 2.2.2) and a limiting fuel storage configuration. The computed temperatures are reported in Table 4.6.1 and the corresponding MPC internal pressure in Table 4.6.7. The results are confirmed to be below the temperature limits (Table 2.2.3) and pressure limit (Table 2.2.1) for off-normal conditions.

#### 4.6.1.4 FHD Malfunction

This event is defined in Subsection 12.1.5 as stoppage of the FHD machine following loss of power or active component trip. The principal effect of this event is stoppage of helium circulation through the MPC and transitioning of heat dissipation in the MPC from forced convection to natural circulation cooling. To bound this event an array of adverse conditions are assumed to have developed coincidentally, as noted below:

- a. Steady state maximum temperatures have been reached.
- b. Design maximum heat load in the limiting MPC-37 is assumed.
- c. Air (not water) is in the HI-TRAC FW annulus.
- d. The helium pressure in the MPC is at the minimum possible value of 20 psig.

Under the FHD malfunction condition the principal requirement to ensure the off-normal cladding temperature limits mandated by ISG-11, Rev. 3 (see Table 2.2.3) must be demonstrated. For this purpose an array of adverse conditions are defined above and the Peak Cladding Temperature (PCT) computed using the 3D FLUENT model of the transfer cask articulated in Section 4.5. The PCT computes as 433°C which is significantly below the 570°C off-normal temperature limit.

#### **4.6.2 Accident Events**

##### **4.6.2.1 Fire Accident (Load Case AB in Table 2.2.13)**

Although the probability of a fire accident affecting a HI-STORM FW system during storage operations is low due to the lack of combustible materials at an ISFSI, a conservative fire event has been assumed and analyzed. The only credible concern is a fire from an on-site transport vehicle fuel tank. Under a postulated fuel tank fire, the outer layers of HI-TRAC VW or HI-STORM FW overpacks are heated for the duration of fire by the incident thermal radiation and forced convection heat fluxes. The amount of fuel in the on-site transporter is limited to a volume of 50 gallons. The data necessary to define the fire event is provided in Table 2.2.8.

##### (a) HI-STORM FW Fire

The fuel tank fire is conservatively assumed to surround the HI-STORM FW overpack. Accordingly, all exposed overpack surfaces are heated by radiation and convection heat transfer from the fire. Based on NUREG-1536 and 10 CFR 71 guidelines [4.6.1], the following fire parameters are assumed:

1. The average emissivity coefficient must be at least 0.9. During the entire duration of the fire, the painted outer surfaces of the overpack are assumed to remain intact, with an emissivity of 0.85. It is conservative to assume that the flame emissivity is 1.0, the limiting maximum value corresponding to a perfect blackbody emitter. With a flame emissivity conservatively assumed to be 1.0 and a painted surface emissivity of 0.85, the effective emissivity coefficient is 0.85. Because the minimum required value of 0.9 is greater than the actual value of 0.85, use of an average emissivity coefficient of 0.9 is conservative.
2. The average flame temperature must be at least 1475°F (802°C). Open pool fires typically involve the entrainment of large amounts of air, resulting in lower average flame temperatures. Additionally, the same temperature is applied to all exposed cask surfaces, which is very conservative considering the size of the HI-STORM FW cask. It

is therefore conservative to use the 1475°F (802°C) temperature.

3. The fuel source must extend horizontally at least 1 m (40 in), but may not extend more than 3 m (10 ft), beyond the external surface of the cask. Use of the minimum ring width of 1 meter yields a deeper pool for a fixed quantity of combustible fuel, thereby conservatively maximizing the fire duration (specified in Table 2.2.8).
4. The convection coefficient must be that value which may be demonstrated to exist if the cask were exposed to the fire specified. Based upon results of large pool fire thermal measurements [4.6.2], a conservative forced convection heat transfer coefficient of 4.5 Btu/(hr×ft<sup>2</sup>×°F) is applied to exposed overpack surfaces during the short-duration fire.

Based on the 50 gallon fuel volume, the overpack outer diameter and the 1 m fuel ring width [4.6.1], the fuel ring surrounding the overpack covers 154.1 ft<sup>2</sup> and has a depth of 0.52 inch. From this depth and the fuel consumption rate of 0.15 in/min, the calculated fire duration is provided in Table 2.2.8. The fuel consumption rate of 0.15 in/min is a lowerbound value from a Sandia National Laboratories report [4.6.2]. Use of a lowerbound fuel consumption rate conservatively maximizes the duration of the fire.

To evaluate the impact of fire heating of the HI-STORM FW overpack, a thermal model of the overpack cylinder was constructed using FLUENT. A transient study is conducted for the duration of fire and post-fire of sufficient duration to reach maximum temperatures. The bounding steady state HI-STORM FW normal storage temperatures (short fuel scenario in MPC-37, see Table 4.4.3) are adopted as the initial condition for the fire accident (fire and post-fire) evaluation. The transient study was conducted for a sufficiently long period to allow temperatures in the overpack to reach their maximum values and begin to recede.

Due to the severity of the fire condition radiative heat flux, heat flux from incident solar radiation is negligible and is not included. Furthermore, the smoke plume from the fire would block most of the solar radiation.

The thermal transient response of the storage overpack is determined using FLUENT. Time-histories for points in the storage overpack are monitored for the duration of the fire and the subsequent post-fire equilibrium phase.

Heat input to the HI-STORM FW overpack while it is subjected to the fire is from a combination of incident radiation and convective heat flux to all external surfaces. This can be expressed by the following equation:

$$q_F = h_{fc} (T_A - T_S) + \sigma \varepsilon [(T_A + C)^4 - (T_S + C)^4]$$

where:

$q_F$  = Surface Heat Input Flux (Btu/ft<sup>2</sup>-hr)

$h_{fc}$  = Forced Convection Heat Transfer Coefficient (4.5 Btu/ft<sup>2</sup>-hr-°F)

$\sigma$  = Stefan-Boltzmann Constant

$T_A$  = Fire Temperature (1475°F)  
 $C$  = Conversion Constant (460 (°F to °R))  
 $T_S$  = Surface Temperature (°F)  
 $\varepsilon$  = Average Emissivity (0.90 per 10 CFR 71.73)

The forced convection heat transfer coefficient is based on the results of large pool fire thermal measurements [4.6.2].

After the fire event, the ambient temperature is restored and the storage overpack cools down (post-fire temperature relaxation). Heat loss from the outer surfaces of the storage overpack is determined by the following equation:

$$q_s = h_s (T_s - T_A) + \sigma \varepsilon [(T_s + C)^4 - (T_A + C)^4]$$

where:

$q_s$  = Surface Heat Loss Flux (W/m<sup>2</sup> (Btu/ft<sup>2</sup>-hr))  
 $h_s$  = Natural Convection Heat Transfer Coefficient (Btu/ft<sup>2</sup>-hr-°F)  
 $T_s$  = Surface Temperature (°F)  
 $T_A$  = Ambient Temperature (°F)  
 $\sigma$  = Stefan-Boltzmann Constant  
 $\varepsilon$  = Surface Emissivity  
 $C$  = Conversion Constant (460 (°F to °R))

In the post-fire temperature relaxation phase,  $h_s$  is obtained using literature correlations for natural convection heat transfer from heated surfaces [4.2.9]. Solar insolation was included during post-fire event. An emissivity of bare carbon steel (see Table 4.2.4) is used for all the cask outer surfaces during post-fire analysis.

The results of the fire and post-fire events are reported in Table 4.6.2. These results demonstrate that the fire accident event has a minor affect on the fuel cladding temperature. Localized regions of concrete upto 1 inch depth are exposed to temperatures in excess of accident temperature limit. The bulk concrete temperature remains below the short-term temperature limit. The temperatures of the basket and components of MPC and HI-STORM FW overpack (see Table 4.6.2) are within the allowable temperature limits.

Table 4.6.2 shows a slight increase in fuel temperature following the fire event. Thus the impact on the MPC internal helium pressure is correspondingly small. Based on a conservative analysis of the HI-STORM FW system response to a hypothetical fire event, it is concluded that the fire event does not adversely affect the temperature of the MPC or contained fuel. Thus, the ability of the HI-STORM FW system to maintain the spent nuclear fuel within design temperature limits during and after fire is assured.

### (b) HI-TRAC VW Fire

In this subsection the fuel cladding and MPC pressure boundary integrity under an exposure to a short duration fire event is demonstrated. The HI-TRAC VW is initially (before fire) assumed to be loaded to design basis decay heat and has reached steady-state maximum temperatures. The analysis assumes a fire from a 50 gallon transporter fuel tank spill. The fuel spill, as discussed in Subsection 4.6.2.1(a) is assumed to surround the HI-TRAC VW in a 1 m wide ring. The fire parameters are same as that assumed for the HI-STORM FW fire discussed in this preceding subsection. In this analysis, the HI-TRAC VW and its contents are conservatively postulated to undergo a transient heat-up as a lumped mass from the decay heat and heat input from the fire.

Based on the specified 50 gallon fuel volume, HI-TRAC VW cylinder diameter (7.9 ft) and the 1 m fuel ring width, the fuel ring area is 115.2 ft<sup>2</sup> and has a depth of 0.696 in. From this depth and the fuel consumption rate of 0.15 in/min, the fire duration  $\tau_f$  is calculated to be 4.64 minutes (279 seconds). The fuel consumption rate of 0.15 in/min is a lowerbound value from Sandia Report [4.6.1]. Use of a lowerbound fuel consumption rate conservatively maximizes the duration of the fire.

From the HI-TRAC VW fire analysis, a bounding rate of temperature rise 2.722°F per minute is determined. Therefore, the total temperature rise is computed as the product of the rate of temperature rise and  $\tau_f$  is 12.6°F. Because the cladding temperature at the start of fire is substantially below the accident temperature limit, the fuel cladding temperature limit during HI-TRAC VW fire is not exceeded. To confirm that the MPC pressure remains below the design accident pressure (Table 2.2.1) the MPC pressure resulting from fire temperature rise is computed using the Ideal Gas Law. The result (see Table 4.6.7) is below the pressure limit (see Table 2.2.1).

#### **4.6.2.2 Jacket Water Loss**

In this subsection, the fuel cladding and MPC boundary integrity is evaluated under a postulated (non-mechanistic) loss of water from the HI-TRAC VW water jacket. For a bounding analysis, all water compartments are assumed to lose their water and be replaced with air. The HI-TRAC VW is assumed to have the maximum thermal payload (design heat load) and assumed to have reached steady state (maximum) temperatures. Under these assumed set of adverse conditions, the maximum temperatures are computed and reported in Table 4.6.3. The results of jacket water loss evaluation confirm that the cladding, MPC and HI-TRAC VW component temperatures are below the limits prescribed in Chapter 2 (Table 2.2.3). The co-incident MPC pressure is also computed and compared with the MPC accident design pressure (Table 2.2.1). The result (Table 4.6.7) shows a positive margin of safety.

#### **4.6.2.3 Extreme Environmental Temperatures**

To evaluate the effect of extreme weather conditions, an extreme ambient temperature (Table 2.2.2) is postulated to persist for a 3-day period. For a conservatively bounding evaluation the

extreme temperature is assumed to last for a sufficient duration to allow the HI-STORM FW system to reach steady state conditions. Because of the large mass of the HI-STORM FW system, with its corresponding large thermal inertia and the limited duration for the extreme temperature, this assumption is conservative. Starting from a baseline condition evaluated in Section 4.4 (normal ambient temperature and limiting fuel storage configuration) the temperatures of the HI-STORM FW system are conservatively assumed to rise by the difference between the extreme and normal ambient temperatures (45°F). The HI-STORM FW extreme ambient temperatures computed in this manner are reported in Table 4.6.4. The co-incident MPC pressure is also computed (Table 4.6.7) and compared with the accident design pressure (Table 2.2.1), which shows a positive safety margin. The result is confirmed to be below the accident limit.

#### **4.6.2.4 100% Blockage of Air Inlets**

This event is defined as a complete blockage of all eight bottom inlets for a significant duration (32 hours). The immediate consequence of a complete blockage of the air inlets is that the normal circulation of air for cooling the MPC is stopped. An amount of heat will continue to be removed by localized air circulation patterns in the overpack annulus and outlet ducts, and the MPC will continue to radiate heat to the relatively cooler storage overpack. As the temperatures of the MPC and its contents rise, the rate of heat rejection will increase correspondingly. Under this condition, the temperatures of the overpack, the MPC and the stored fuel assemblies will rise as a function of time.

As a result of the considerable inertia of the storage overpack, a significant temperature rise is possible if the inlets are substantially blocked for extended durations. This accident condition is, however, a short duration event that is identified and corrected through scheduled periodic surveillance. Nevertheless, this event is conservatively analyzed assuming a substantial duration of blockage. The HI-STORM FW thermal model is the same 3-Dimensional model constructed for normal storage conditions (see Section 4.4) except for the bottom inlet ducts, which are assumed to be impervious to air. Using this model, a transient thermal solution of the HI-STORM FW system starting from normal storage conditions is obtained. The results of the blocked ducts transient analysis are presented in Table 4.6.5 and compared against the accident temperature limits (Table 2.2.3). The co-incident MPC pressure (Table 4.6.7) is also computed and compared with the accident design pressure (Table 2.2.1). All computed results are well below their respective limits.

#### **4.6.2.5 Burial Under Debris (Load Case AG in Table 2.2.13)**

Burial of the HI-STORM FW system under debris is not a credible accident. During storage at the ISFSI there are no structures that loom over the casks whose collapse could completely bury the casks in debris. Minimum regulatory distances from the ISFSI to the nearest ISFSI security fence precludes the close proximity of substantial amount of vegetation. There is no credible



mechanism for the HI-STORM FW system to become completely buried under debris. However, for conservatism, the scenario of complete burial under debris is considered.

For this purpose, an exceedingly conservative analysis that considers the debris to act as a perfect insulator is considered. Under this scenario, the contents of the HI-STORM FW system will undergo a transient heat up under adiabatic conditions. The minimum available time ( $\Delta\tau$ ) for the fuel cladding to reach the accident limit depends on the following: (i) thermal inertia of the cask, (ii) the cask initial conditions, (iii) the spent nuclear fuel decay heat generation and (iv) the margin between the initial cladding temperature and the accident temperature limit. To obtain a lowerbound on  $\Delta\tau$ , the HI-STORM FW overpack thermal inertia (item i) is understated, the cask initial temperature (item ii) is maximized, decay heat overstated (item iii) and the cladding temperature margin (item iv) is understated. A set of conservatively postulated input parameters for items (i) through (iv) are summarized in Table 4.6.6. Using these parameters  $\Delta\tau$  is computed as follows:

$$\Delta\tau = \frac{m \times c_p \times \Delta T}{Q}$$

where:

- $\Delta\tau$  = minimum available burial time (hr)
- $m$  = Mass of HI-STORM FW System (lb)
- $c_p$  = Specific heat capacity (Btu/lb-°F)
- $\Delta T$  = Permissible temperature rise (°F)
- $Q$  = Decay heat load (Btu/hr)

Substituting the parameters in Table 4.6.6, the minimum available burial time is computed as 57.6 hours for the short fuel assembly (15x15I). A site-specific calculation based on the methodology described herein can be performed to determine the burial time limits. The coincident MPC pressure (see Table 4.6.7) is also computed and compared with the accident design pressure (Table 2.2.1). These results indicate that HI-STORM FW has a substantial thermal sink capacity to withstand complete burial-under-debris events.

#### 4.6.2.6 Evaluation of Smart Flood (Load Case AD in Table 2.2.13)

A number of design measures are taken in the HI-STORM FW system to limit the fuel cladding temperature rise under a most adverse flood event (i.e., one that is just high enough to block the inlet duct). An unlikely adverse flood accident is assumed to occur with flood water up to the inlet height and is termed as ‘smart flood’. The inlet duct is narrow and tall so that blocking the inlet ducts completely would require that flood waters wet the bottom region of the MPC creating a heat sink.

The inlet duct is configured to block radiation efficiently even if the radiation emanating from the MPC is level (coplanar) with the duct penetration. The MPC stands on the base plate, which is welded to the inner and outer shell of the overpack. Thus, if the flood water rises high enough

to block air flow through the bottom ducts, the lower region of the MPC will be submerged in the water. Although heat transport through air circulation is cut off in this scenario, the reduction is substantially offset by flood water cooling.

The MPCs are equipped with the thermosiphon capability, which brings the heat emitted by the fuel to the bottom region of the MPC as the circulating helium flows along the downcomer space around the basket. This places the heated helium in close thermal communication with the flood water, further enhancing convective cooling via the flood water.

The most adverse flood condition exists when the flood waters are high enough to block the inlet ducts but no higher. In this scenario, the MPC surface has minimum submergence in water and the ventilation air is completely blocked. In fact, as the flood water begins to accumulate on the ISFSI pad, the air passage size in the inlet vents is progressively reduced. Therefore, the rate of floodwater rise with time is necessary to analyze the thermal-hydraulic problem. For the reference design basis flood (DBF) analysis in this FSAR, the flood waters are assumed to rise instantaneously to the height to block the inlet vents and stay at that elevation for 32 hours. The consequences of the DBF event is bounded by the 100% blocked ducts events evaluated in Section 4.6.2.4. If the duration of the flood blockage exceeds the DBF blockage duration then a site specific evaluation shall be performed in accordance with the methodology presented in this Chapter and evaluated for compliance with Subsection 2.2.3 criteria.

Table 4.6.1		
OFF-NORMAL CONDITION MAXIMUM HI-STORM FW TEMPERATURES		
Component	Off-Normal Ambient Temperature °C (°F)	Partial Inlets Duct Blockage °C (°F)
Fuel Cladding	384 (723)	385 (725)
MPC Basket	369 (696)	371 (700)
Aluminum Basket Shims	301 (574)	285 (545)
MPC Shell	251 (484)	257 (495)
MPC Lid1	246 (475)	252 (486)
Overpack Inner Shell	137 (279)	141 (286)
Overpack Outer Shell	76 (169)	62 (144)
Overpack Body Concrete <sup>*</sup>	100 (212)	95 (203)
Overpack Lid Concrete <sup>*</sup>	122 (252)	122 (252)

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1 Obtained by adding the difference between extreme ambient and normal temperature difference (11.1°C (20°F)) to normal condition temperatures reported in Table 4.4.3.

<p>Table 4.6.2</p> <p>HI-STORM FW FIRE AND POST-FIRE ACCIDENT ANALYSIS RESULTS</p>			
<b>Component</b>	<b>Initial Condition<sup>Note 3</sup></b> <b>°C (°F)</b>	<b>End of Fire Condition</b> <b>°C (°F)</b>	<b>Post-Fire Cooldown</b> <b>°C (°F)</b>
Fuel Cladding	375 (707)	375 (707)	376 (709)
MPC Basket	361 (682)	361 (682)	362 (684)
Basket Periphery	297 (567)	297 (567)	298 (568)
Aluminum Basket Shims	276 (529)	276 (529)	277 (531)
MPC Shell	246 (475)	250 (482)	250 (482)
MPC Lid <sup>Note 1</sup>	243 (469)	242 (468)	244 (471)
Overpack Inner Shell	128 (262)	133 (271)	133 (271)
Overpack Outer Shell	60 (140)	337 (639) <sup>Note 2</sup>	337 (639) <sup>Note 2</sup>
Overpack Body Concrete <sup>Note 1</sup>	88 (190)	93 (199)	93 (199)
Overpack Lid Concrete <sup>Note 1</sup>	113 (235)	121 (250)	121 (250)
<p>Note 1: Maximum section average temperature is reported.</p> <p>Note 2: Surface average temperature is reported.</p> <p>Note 3: The initial condition used for the thermal analysis of hypothetical fire condition bounds the predicted temperatures for normal long-term storage condition (Table 4.4.3).</p>			

Table 4.6.3 HI-TRAC VW JACKET WATER LOSS MAXIMUM TEMPERATURES	
Component	Temperature °C (°F)
Fuel Cladding	432 (810)
MPC Basket	416 (781)
Basket Periphery	342 (648)
Aluminum Basket Shims	314 (597)
MPC Shell	290 (554)
MPC Lid <sup>1</sup>	263 (505)
HI-TRAC VW Inner Shell	205 (401)
HI-TRAC VW Radial Lead Gamma Shield	204 (399)

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<sup>1</sup> Maximum section average temperature is reported.

Table 4.6.4 EXTREME ENVIRONMENTAL CONDITION MAXIMUM HI-STORM FW TEMPERATURES	
Component	Temperature <sup>1</sup> °C (°F)
Fuel Cladding	398 (748)
MPC Basket	383 (721)
Basket Periphery	315 (599)
Aluminum Basket Shims	292 (558)
MPC Shell	265 (509)
MPC Lid <sup>Note 1</sup>	260 (500)
Overpack Inner Shell	151 (304)
Overpack Outer Shell	90 (194)
Overpack Body Concrete <sup>Note 1</sup>	114 (237)
Overpack Lid Concrete <sup>Note 1</sup>	136 (277)
Average Air Outlet	128 (262)
Note 1: Maximum section average temperature is reported.	

1 Obtained by adding the difference between extreme ambient and normal temperature difference (25°C (45°F)) to normal condition temperatures reported in Table 4.4.3.

Table 4.6.5	
RESULTS OF HI-STORM FW 32-HOURS BLOCKED INLET DUCTS THERMAL ANALYSIS	
Component	Temperature °C (°F)
Fuel Cladding	470 (878)
MPC Basket	453 (847)
Basket Periphery	388 (730)
Aluminum Basket Shims	363 (685)
MPC Shell	340 (644)
MPC Lid <sup>Note 1</sup>	293 (559)
Overpack Inner Shell	238 (460)
Overpack Outer Shell	105 (221)
Overpack Body Concrete <sup>Note 1</sup>	123 (253)
Overpack Lid Concrete <sup>Note 1</sup>	154 (309)
Note 1: Maximum section average temperature is reported.	

Table 4.6.6	
SUMMARY OF INPUTS FOR BURIAL UNDER DEBRIS ANALYSIS	
Thermal Inertia Inputs <sup>1</sup> :	
M (Lowerbound HI-STORM FW Weight)	139172 kg
Cp (Carbon steel heat capacity) <sup>2</sup>	419 J/kg-°C
Clad initial temperature <sup>Note 1</sup>	390°C
Q (Decay heat)	45 kW
$\Delta T$ (clad temperature margin) <sup>3</sup>	160°C
Note 1: Initial temperature conservatively postulated to bound the maximum cladding temperature.	

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1 Thermal inertia of fuel is conservatively neglected.

2 Used carbon steel's specific heat since it has the lowest heat capacity among the principal materials employed in MPC and overpack construction (carbon steel, stainless steel, Metamic-HT and concrete).

3 The clad temperature margin is conservatively understated in this table.



Table 4.6.7	
OFF-NORMAL AND ACCIDENT CONDITION MAXIMUM MPC PRESSURES	
Condition	Pressure (psig)
Off-Normal Conditions	
Off-Normal Pressure <sup>1</sup>	110.0
Partial Blockage of Inlet Ducts	99.9
Accident Conditions	
HI-TRAC VW fire accident	103.3
Extreme Ambient Temperature	101.7
100% Blockage of Air Inlets	116.4
Burial Under Debris	130.8
HI-TRAC VW Jacket Water Loss	109.5

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<sup>1</sup> The off-normal pressure event defined in Section 4.6.1.1 bounds the off-normal ambient temperature event (Section 4.6.1.2)

## 4.7 REGULATORY COMPLIANCE

### 4.7.1 Normal Conditions of Storage

NUREG-1536 [4.4.1] and ISG-11 [4.1.4] define several thermal acceptance criteria that must be applied to evaluations of normal conditions of storage. These items are addressed in Sections 4.1 through 4.4. Each of the pertinent criteria and the conclusion of the evaluations are summarized here.

As required by ISG-11 [4.1.4], the fuel cladding temperature at the beginning of dry cask storage is maintained below the anticipated damage-threshold temperatures for normal conditions for the licensed life of the HI-STORM FW System. Maximum clad temperatures for long-term storage conditions are reported in Section 4.4.

As required by NUREG-1536 (4.0,IV,3), the maximum internal pressure of the cask remains within its design pressure for normal conditions, assuming rupture of 1 percent of the fuel rods. 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 Section 4.4 and shown to remain below the normal design pressures specified in Table 2.2.1.

As required by NUREG-1536 (4.0,IV,4), all cask and fuel materials are maintained within their minimum and maximum temperature for normal and off-normal conditions in order to enable components to perform their intended safety functions. Maximum and minimum temperatures for long-term storage conditions are reported in Section 4.4 which are shown to be well below their respective Design temperature limits summarized in Table 2.2.3.

As required by NUREG-1536 (4.0,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 shown to be below the ISG-11 [4.1.4] limit of 400°C (752°F).

As required by NUREG-1536 (4.0,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.

As required by NUREG-1536 (4.0,IV,8), the thermal performance of the cask is within the allowable design criteria specified in SAR Chapters 2 and 3 for normal conditions. All thermal results reported in Section 4.4 are within the design criteria under all normal conditions of storage.

### 4.7.2 Short-Term Operations

Evaluation of short-term operations is presented in Section 4.5 wherein complete compliance with the provisions of ISG-11 [4.1.4] is demonstrated. In particular, the ISG-11 requirement to

ensure that maximum cladding temperatures under all fuel loading and short-term operations be below 400°C (752°F) for high burnup fuel and below 570°C (1058°F) for moderate burnup fuel (Table 4.3.1) is demonstrated.

Further, as required by NUREG-1536 (4.0,IV, 4), all cask and fuel materials are maintained within their minimum and maximum temperature for all short-term operations in order to enable components to perform their intended safety functions.

As required by NUREG-1536 (4.0,IV,8), the thermal performance of the cask is within the allowable design criteria specified in SAR Chapters 2 and 3 for all short-term operations.

#### **4.7.3 Off-Normal and Accident Conditions**

As required by NUREG-1536 (4.0,IV,3), the maximum internal pressure of the cask is evaluated in Section 4.6 and shown to remain within its off-normal and accident design pressure, assuming rupture of 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.

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## CHAPTER 5<sup>†</sup>: SHIELDING EVALUATION

### 5.0 INTRODUCTION

The shielding analysis of the HI-STORM FW system is presented in this chapter. As described in Chapter 1, the HI-STORM FW system is designed to accommodate both PWR and BWR MPCs within HI-STORM FW overpacks (see Table 1.0.1).

In addition to storing intact PWR and BWR fuel assemblies, the HI-STORM FW system is designed to store BWR and PWR damaged fuel assemblies and fuel debris. Damaged fuel assemblies and fuel debris are defined in Subsection 2.1. Both damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs).

PWR fuel assemblies may contain burnable poison rod assemblies (BPRAs), with any number of full-length rods and thimble plug rodlets in the locations without a full-length rod, thimble plug devices (TPDs), control rod assemblies (CRAs) or axial power shaping rod assemblies (APSRs), neutron source assemblies (NSAs), or similarly named devices. These non-fuel hardware devices are an integral yet removable part of PWR fuel assemblies and therefore the HI-STORM FW system has been designed to store PWR fuel assemblies with or without these devices. Since each device occupies the same location within a fuel assembly, a single PWR fuel assembly will not contain multiple devices, with the exception of instrument tube tie rods (ITTRs), which may be stored in the assembly along with other types of non-fuel hardware.

As described in Chapter 1 (see Tables 1.2.3 and 1.2.4), the loading of fuel in all HI-STORM FW MPCs will follow specific heat load limitations.

In order to offer the user more flexibility in fuel storage, the HI-STORM FW System offers two heat load patterns, each with a three-region loading configuration, in the MPC-37. The MPC-89 has one heat load pattern with a three-region loading configuration. The regionalized storage patterns are guided by the considerations of minimizing occupational and site boundary dose to comply with ALARA principles.

The sections that follow will demonstrate that the design of the HI-STORM FW dry cask storage system fulfills the following acceptance criteria outlined in the Standard Review Plan, NUREG-1536 [5.2.1]:

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<sup>†</sup> 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. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary and component nomenclature of the Bill-of-Materials (Section 1.5).

### 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 system designer 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 10CFR72.106.
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 10CFR Part 20, Subparts C and D.

Consistent with the Standard Review Plan, NUREG-1536, this chapter contains the following information:

- A description of the shielding features of the HI-STORM FW system, including the HI-TRAC transfer cask.
- A description of the source terms.
- A general description of the shielding analysis methodology.
- A description of the analysis assumptions and results for the HI-STORM FW system, including the HI-TRAC transfer cask.
- Analyses are presented for each MPC showing that the radiation dose rates follow As-Low-As-Reasonably-Achievable (ALARA) practices.
- Analyses to show that the 10CFR72.106 controlled area boundary radiation dose limits can be met during accident conditions of storage for non-effluent radiation from illustrative ISFSI configurations at a minimum distance of 100 meters. Since only representative dose rate values for normal conditions are presented for this chapter, compliance with the



radiation and exposure objectives of 10CFR72.104 is not being evaluated herein but will be performed as part of the site specific evaluations.

Chapter 2 contains a detailed description of structures, systems, and components important to safety.

Chapter 7 contains a discussion on the release of radioactive materials from the HI-STORM FW system. Therefore, this chapter only calculates the dose from direct neutron and gamma radiation emanating from the HI-STORM FW system.

Chapter 11, Radiation Protection, contains the following information:

- A discussion of the estimated occupational exposures for the HI-STORM FW system, including the HI-TRAC transfer cask.
- A summary of the estimated radiation exposure to the public.

The safety analyses summarized in this chapter demonstrate that under accident conditions, acceptable margins to allowable limits exist under all design basis loading conditions. For normal and off-normal conditions, the analyses in this chapter simply provide a generic evaluation that demonstrates that the dose requirements as specified in 10CFR72.104 can be met under site specific conditions. 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 dose rates in this chapter 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 unquantified changes cannot be deemed any more to be insignificant. For accident conditions, the dose limit as specified in 10CFR72.106 is 5 rem. The only accident which impacts dose rates is the loss of water in the water jacket for the HI-TRAC VW. For the purposes of determining if the changes to the HI-TRAC VW are insignificant, an insignificant loss of margin with reference to the 5 rem acceptance criteria is defined as the estimated reduction that is no more than one order of magnitude less than the available margin reported in the FSAR. For normal and off-normal conditions, site specific dose evaluations are required to demonstrate compliance with 10CFR72.104. Incorporating any minor changes into those site specific evaluations is only warranted if it would be expected, on a site specific basis, that those changes could result in a situation where the limits are no longer met and where therefore other compensatory measures are required, such as a change in the loading plan or the concrete density. Incorporating changes into the analyses in this chapter for normal and off-normal conditions will only be performed under extenuating circumstances, e.g. major changes to the shielding design, in order to provide an updated template for the site specific dose analyses.

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.

## 5.1 DISCUSSION AND RESULTS

The principal sources of radiation in the HI-STORM FW 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.  $\alpha, n$  reactions in fuel materials
  3. Secondary neutrons produced by fission from subcritical multiplication
  4.  $\gamma, n$  reactions (this source is negligible)

During loading, unloading, and transfer operations, shielding from gamma radiation is provided by the stainless steel structure and the basket of the MPC and the steel, lead, and water in the HI-TRAC transfer cask. For storage, the gamma shielding is provided by the MPC, and the steel and concrete (“Metcon” structure) of the overpack. Shielding from neutron radiation is provided by the concrete of the overpack during storage and by the water of the HI-TRAC transfer cask during loading, unloading, and transfer operations. It is worth noting that the models, used to evaluate the dose calculations in this chapter, are constructed with minimum concrete densities and minimum lead thicknesses.

The shielding analyses were performed with MCNP5 [5.1.1] developed by Los Alamos National Laboratory (LANL). The source terms for the design basis fuels were calculated with the SAS2H and ORIGEN-S sequences from the SCALE 5 system [5.1.2, 5.1.3]. A detailed description of the MCNP models and the source term calculations are presented in Sections 5.3 and 5.2, respectively.

The design basis zircaloy clad fuel assemblies used for calculating the dose rates presented in this chapter are Westinghouse (W) 17x17 and the General Electric (GE) 10x10, for PWR and BWR fuel types, respectively. Required site specific shielding evaluations will verify whether those assemblies and assembly parameters are appropriate for the site-specific analyses. Subsection 2.1 specifies the acceptable fuel characteristics, including the acceptable maximum burnup levels and minimum cooling times for storage of fuel in the HI-STORM FW MPCs.

The following presents a discussion that explains the rationale behind the burnup and cooling time combinations that are evaluated in this chapter for normal and accident conditions.

10CFR72 contains two sections that set down main dose rate requirements: §104 for normal and off-normal conditions, and §106 for accident conditions. The relationship of these requirements

to the analyses in this Chapter 5, and the burnup and cooling times selected for the various analyses, are as follows:

- 10CFR72.104 specifies the dose limits from an ISFSI (and other operations) at a site boundary under normal and off-normal conditions. Compliance with §104 can therefore only be demonstrated on a site-specific basis, since it depends not only on the design of the cask system and the loaded fuel, but also on the ISFSI layout, the distance to the site boundary, and possibly other factors such as use of higher density concrete or the terrain around the ISFSI. The purpose of this chapter is therefore to present a general overview over the expected dose rates, next to the casks and at various distances, to aid the user in applying ALARA considerations and planning of the ISFSI. To that extent, it is sufficient to present reasonably conservative dose rate values, based on a reasonable conservative choice of burnups and cooling times of the assemblies.
- For the accident dose limit in 10CFR72.106 it is desirable to show compliance in this Chapter 5 on a generic basis, so that calculations on a site-by-site basis are not required. To that extent, a burnup and cooling time calculation that maximizes the dose rate under accident conditions needs to be selected.

The HI-STORM FW System offers three-region loading configurations as shown in Table 1.2.3 and Table 1.2.4 in Chapter 1.

- For the MPC-37, there are two heat load patterns, each with a three-region loading configuration – Loading Pattern A and Loading Pattern B. An important difference between Pattern A and Pattern B loading is the loading is the maximum allowed heat load of the cells on the periphery of the MPC-37. Pattern A contains the cells with the lowest decay heat on the periphery, while Pattern B contains the cells with the highest decay heat on the periphery. In Pattern A, fuel assemblies with higher heat loads are loaded in the inner region allowing the user to take advantage of self-shielding from the fuel assemblies with lower heat loads in the outer regions. However, for Pattern B, the fuel assemblies with the higher heat loads could be loaded in the outer region (Region 3). Based on this difference it is expected that Pattern B will have higher dose rates than Pattern A. Therefore, for dose calculations Pattern B is selected, as it is the more limiting of the two loading patterns. Furthermore, uniform loading of MPC-37 cells is assumed for dose calculations. The burnup and cooling time combination is selected as representative of the cells on the periphery. This is a conservative approach, as it assumes that all thirty seven cells have a decay heat per cell equal to or slightly exceeding the decay heat of the periphery cells.
- For the MPC-89, there is only one heat load pattern with a three-region loading configuration. Based on the configuration for the MPC-89, fuel assemblies with higher heat loads would be loaded in the inner region allowing the user to take advantage of self-shielding from fuel assemblies with lower heat loads in the outer regions (see Table 1.2.4). However, for simplification, the shielding analyses are performed for a single region, i.e. assuming all assemblies in the basket have the same burnup and cooling time. In the case of the MPC-89 the burnup and cooling time combination is selected as a representative average for the entire basket.

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While Loading Pattern B for the MPC-37 allows assemblies with higher heat loads and therefore higher source terms in the outer region (Region 3) of the MPC, the guiding principle in selecting fuel loading should still be to preferentially place assemblies with higher source terms in the inner regions of the basket as far as reasonably possible.

It is recognized that for a given heat load, an infinite number of burnup and cooling time combination could be selected, which would result in slightly different dose rate distributions around the cask. For a high burnup with a corresponding longer cooling time, dose locations with a high neutron contribution would show higher dose values, due to the non-linear relationship between burnup and neutron source term. At other locations dose rates are more dominated by contribution from the gamma sources. In these cases, short cooling time and lower burnup combinations with heat load comparable to the higher burnup and corresponding longer cooling time combinations would result in higher dose rates. However, in those cases, there would always be a compensatory effect, since for each dose location, higher neutron dose rates would be partly offset by lower gamma dose rates and vice versa.

Based on these considerations, average burnup and cooling time values are selected for all calculations for normal conditions, i.e values that are away from the extreme values. The selected values are shown in Table 5.0.1, and are based on a total heat loads presented in Table 1.2.3. For the accident conditions however, it is recognized that the bounding accident condition is the loss of water in the HI-TRAC VW, a condition that is neutron dominated due to the removal of the principal neutron absorber in the HI-TRAC VW (water). For this case, the upper bound burnup is selected, in order to maximize the neutron source strength of all assemblies in the basket, and a corresponding higher cooling time is selected in order to meet the overall heat load limit in the cask. The resulting burnup and cooling times values for accidents are therefore different from those for normal conditions and are listed in Table 5.0.2. In all cases, low initial enrichments are selected, which further increases the neutron source terms from the assemblies

With the burnup and cooling times selected based on above considerations, dose rates calculated for normal conditions will be reasonably conservative, while for accident conditions those will represent reasonable upper bound limits.

Table 5.0.1

DESIGN BASIS FUEL BURNUP, COOLING TIME AND ENRICHMENT FOR NORMAL  
CONDITIONS

Design Basis Burnup and Cooling Times Zircaloy Clad Fuel	
<b>MPC-37</b>	<b>MPC-89</b>
45,000 MWD/MTU 4.5 Year Cooling 3.6 wt% U-235 Enrichment	45,000 MWD/MTU 5 Year Cooling 3.2 wt% U-235 Enrichment

Table 5.0.2

DESIGN BASIS FUEL BURNUP, COOLING TIME AND ENRICHMENT FOR ACCIDENT  
CONDITIONS

Design Basis Burnup and Cooling Times Zircaloy Clad Fuel	
<b>MPC-37</b>	<b>MPC-89</b>
65,000 MWD/MTU 8 Year Cooling 4.8 wt% U-235 Enrichment	65,000 MWD/MTU 10 Year Cooling 4.8 wt% U-235 Enrichment

### 5.1.1 Normal and Off-Normal Operations

Chapter 12 discusses the potential off-normal conditions and their effect on the HI-STORM FW 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 to the whole body, 75 mrem to the thyroid and 25 mrem to any other critical organ.
2. Operational restrictions must be established to meet as low as reasonably achievable (ALARA) 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 11 specifically addresses these regulations.

In accordance with ALARA practices, design objective dose rates are established for the HI-STORM FW system and presented in Table 2.3.2.

Figure 5.1.1 identifies the locations of the dose points referenced in the dose rate summary tables for the HI-STORM FW overpack. Dose Point #2 is located on the side of the cask at the axial mid-height. Dose Points #1 and #3 are the locations of the inlet and outlet air ducts, respectively. The dose values reported for these locations (adjacent and 1 meter) were averaged over the duct opening. Dose Point #4 is the dose location on the overpack lid. 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.

Figure 5.1.2 identifies the location of the dose points for the HI-TRAC VW transfer cask. Dose Point Locations #1 and #3 are situated below and above the water jacket, respectively. Dose Point #4 is the dose location on the HI-TRAC VW lid and dose rates below the HI-TRAC VW are estimated with Dose Point #5. Dose Point Location #2 is situated on the side of the cask at the axial mid-height.

The total dose rates presented are presented for two cases: with and without BPRAs. The dose from the BPRAs was conservatively assumed to be the maximum calculated in Subsection 5.2.4.

Tables 5.1.1 and 5.1.2 provides dose rates adjacent to and one meter from the HI-TRAC VW during normal conditions for the MPC-37 and MPC-89. The dose rates listed in Table 5.1.1 correspond to the normal condition in which the MPC is dry and the HI-TRAC water jacket is filled with water.

Tables 5.1.5 and 5.1.6 provide the design basis dose rates adjacent to the HI-STORM FW overpack during normal conditions for the MPC-37 and MPC-89. Tables 5.1.7 and 5.1.8 provide the design basis dose rates at one meter from the HI-STORM FW overpack containing the MPC-37 and MPC-89, respectively.

The dose to any real individual at or beyond the controlled area boundary is required to be below 25 mrem per year. The minimum distance to the controlled area boundary is 100 meters from the ISFSI. Table 5.1.3<sup>[SG1]</sup> presents the annual dose to an individual from a single HI-STORM FW cask and various storage cask arrays, assuming an 8760 hour annual occupancy at the dose point location. The minimum distance required for the corresponding dose is also listed. It is noted that these data are provided for illustrative purposes only. A detailed site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10CFR72.212. The site-specific evaluation will consider dose from other portions of the facility and will consider the actual conditions of the fuel being stored (burnup and cooling time).

Figure 5.1.3 is an annual dose versus distance graph for the HI-STORM FW cask array configurations provided in Table 5.1.3. This curve, which is based on an 8760 hour occupancy, is provided for illustrative purposes only and will be re-evaluated on a site-specific basis.

Subsection 5.2.3 discusses the BPRAs, TPDs, CRAs and APSRs that are permitted for storage in the HI-STORM FW system. Subsection 5.4.4 discusses the increase in dose rate as a result of adding non-fuel hardware in the MPCs.

The analyses summarized in this section demonstrate that the HI-STORM FW system is in compliance with the radiation and exposure objectives of 10CFR72.106. Since only representative dose rate values for normal conditions are presented in this chapter, compliance with 10CFR72.104 is not being evaluated. This will be performed as part of the site specific evaluations.

## 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, 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. The lens dose equivalent shall not exceed 15 Rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 Rem. 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.



Structural evaluations, presented in Chapter 3, shows that a freestanding HI-STORM FW storage overpack containing a loaded MPC remains standing during events that could potentially lead to a tip-over event. Therefore, the tip-over accident is not considered as part of the shielding evaluation.

Design basis accidents which may affect the HI-STORM FW overpack can result in limited and localized damage to the outer shell and radial concrete shield. As the damage is localized and the vast majority of the shielding material remains intact, the effect on the dose at the site boundary is negligible. Therefore, the site boundary doses for the loaded HI-STORM FW overpack for accident conditions are equivalent to the normal condition doses, which meet the 10CFR72.106 radiation dose limits. However the adjacent and one meter dose rates may be increased, which should be considered in any post-accident activities near the affected cask.

The design basis accidents analyzed in Chapter 11 have one bounding consequence that affects the shielding materials of the HI-TRAC transfer cask. It is the potential for damage to the water jacket shell and the loss of the neutron shield (water). In the accident consequence analysis, it is conservatively assumed that the neutron shield (water) is completely lost and replaced by a void.

Throughout all design basis accident conditions the axial location of the fuel will remain fixed within the MPC because of the MPC's design features (see Chapter 1). Further, the structural evaluation of the HI-TRAC VW in Chapter 3 shows that the inner shell, lead, and outer shell remain intact throughout all design basis accident conditions. Localized damage of the HI-TRAC outer shell is possible; however, localized deformations will have only a negligible impact on the dose rate at the boundary of the controlled area.

The complete loss of the HI-TRAC neutron shield significantly affects the dose at mid-height (Dose Point #2) adjacent to the HI-TRAC. Loss of the neutron shield has a small effect on the dose at the other dose points. To illustrate the impact of the design basis accident, the dose rates at Dose Point #2 (see Figure 5.1.2) are provided in Table 5.1.4 (MPC-37) for the HI-TRAC VW at a distance of 1 meter and at a distance of 100 meters. The normal condition dose rates are provided for reference. The dose for a period of 30 days is shown in Table 5.1.9, where 30 days is used to illustrate the radiological impact for a design basis accident. Based on this dose rate and the short duration of use for the loaded HI-TRAC transfer cask, it is evident that the dose as a result of the design basis accident cannot exceed 5 rem at the controlled area boundary for the short duration of the accident.

Analyses summarized in this section demonstrate that the HI-STORM FW system, including the HI-TRAC VW transfer cask, is in compliance with the 10CFR72.106 limits.

<p style="text-align: center;">Table 5.1.1</p> <p style="text-align: center;">DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS</p> <p style="text-align: center;">MPC-37 DESIGN BASIS FUEL</p> <p style="text-align: center;">45,000 MWD/MTU AND 4.5-YEAR COOLING</p>						
Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	$^{60}\text{Co}$ Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE HI-TRAC VW</b>						
1	975	25	808	67	1874	1874
2	2939	75	<1	154	3169	3169
3	20	5	339	6	371	561
4	98	1	530	225	854	1147
5	940	3	2074	1022	4038	4038
<b>ONE METER FROM THE HI-TRAC VW</b>						
1	695	12	99	30	835	835
2	1382	22	10	58	1472	1474
3	268	6	142	9	425	501
4	80	<1	295	73	449	613
5	470	1	1129	297	1897	1897

## Notes:

- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- Streaming may occur through the annulus. However, during handling/operations the annulus is filled with water and lead snakes are typically present to reduce the streaming effects. Further, operators are not present on top of the transfer cask.
- The “Fuel Gammas” category includes gammas from the spent fuel,  $^{60}\text{Co}$  from the spacer grids, and  $^{60}\text{Co}$  from the BPRAs in the active fuel region.

<p style="text-align: center;">Table 5.1.2</p> <p style="text-align: center;">DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS</p> <p style="text-align: center;">MPC-89 DESIGN BASIS FUEL</p> <p style="text-align: center;">45,000 MWD/MTU AND 5-YEAR COOLING</p>					
Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	$^{60}\text{Co}$ Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
<b>ADJACENT TO THE HI-TRAC VW</b>					
1	244	18	2247	40	2549
2	2466	107	<1	219	2793
3	3	3	581	4	591
4	25	<1	505	138	669
5	132	2	2135	720	2989
<b>ONE METER FROM THE HI-TRAC VW</b>					
1	411	13	291	29	744
2	1142	30	21	74	1267
3	119	5	280	8	412
4	16	<1	300	43	360
5	79	<1	1202	202	1484

## Notes:

- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- Streaming may occur through the annulus. However, during handling/operations the annulus is filled with water and lead snakes are typically present to reduce the streaming effects. Further, operators are not present on top of the transfer cask.
- The “Fuel Gammas” category includes gammas from the spent fuel and  $^{60}\text{Co}$  from the spacer grids.

Table 5.1.3

## DOSE RATES FOR ARRAYS OF HI-STORM FWs with MPC-37

Array Configuration	1 cask	2x2	2x3	2x4	2x5
<b>HI-STORM FW Overpack</b>					
<b>45,000 MWD/MTU AND 4.5-YEAR COOLING</b>					
Annual Dose (mrem/year)	18	15	23	11	14
Distance to Controlled Area Boundary (meters)	300	400	400	500	500

## Notes:

- Values are rounded to nearest integer.
- 8760 hour annual occupancy is assumed.
- Dose location is at the center of the long side of the array.

Table 5.1.4

DOSE RATES FROM HI-TRAC VW WITH MPC-37  
FOR ACCIDENT CONDITIONS  
AT BOUNDING BURNUP AND COOLING TIMES

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	$^{60}\text{Co}$ Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ONE METER FROM HI-TRAC VW</b>						
<b>65,000 MWD/MTU AND 8-YEAR COOLING</b>						
2 (Accident Condition)	1735	3	13	2651	4403	4407
2 (Normal Condition)	893	50	7	122	1071	1074
<b>100 METERS FROM HI-TRAC VW</b>						
<b>65,000 MWD/MTU AND 8-YEAR COOLING</b>						
2 (Accident Condition)	0.7	<0.1	0.1	1.4	2.3	2.4

## Notes:

- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer where appropriate.
- The “Fuel Gammas” category includes gammas from the spent fuel,  $^{60}\text{Co}$  from the spacer grids, and  $^{60}\text{Co}$  from the BPRAs in the active fuel region.

Table 5.1.5

DOSE RATES ADJACENT TO HI-STORM FW OVERPACK  
FOR NORMAL CONDITIONS  
MPC-37  
BURNUP AND COOLING TIME  
45,000 MWD/MTU AND 4.5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	$^{60}\text{Co}$ Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	273	2	14	4	292	292
2	135	1	<1	1	141	141
3 (surface)	11	1	25	2	39	53
3 (overpack edge)	13	<1	63	1	78	113
4 (center)	<1	1	<1	<1	<4	<4
4 (mid)	1	1	4	1	7	10
4 (outer)	10	<1	30	<1	42	59

## Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- Dose location 3 (surface) is at the surface of the outlet vent. Dose location 3 (overpack edge) is in front of the outlet vent, but located radially above the overpack outer diameter.
- Dose location 4 (center) is at the center of the top surface of the top lid. Dose location 4 (mid) is situated directly above the vertical section of the outlet vent. Dose location 4 (outer) is extended along the top plane of the top lid, located radially above the overpack outer diameter.
- The “Fuel Gammas” category includes gammas from the spent fuel,  $^{60}\text{Co}$  from the spacer grids, and  $^{60}\text{Co}$  from the BPRAs in the active fuel region.

Table 5.1.6

DOSE RATES ADJACENT TO HI-STORM FW OVERPACK  
FOR NORMAL CONDITIONS  
MPC-89  
BURNUP AND COOLING TIME  
45,000 MWD/MTU AND 5-YEAR COOLING

<b>Dose Point Location</b>	<b>Fuel Gammas (mrem/hr)</b>	<b>(n,<math>\gamma</math>) Gammas (mrem/hr)</b>	<b><math>^{60}\text{Co}</math> Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>
1	172	2	31	3	208
2	92	2	<1	1	96
3 (surface)	3	<1	29	2	35
3 (overpack edge)	5	<1	69	<1	76
4 (center)	0.1	0.4	0.4	0.1	1
4 (mid)	0.2	0.5	4.3	0.5	6
4 (outer)	2	<1	33	<1	37

## Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- Dose location 3 (surface) is at the surface of the outlet vent. Dose location 3 (overpack edge) is in front of the outlet vent, but located radially above the overpack outer diameter.
- Dose location 4 (center) is at the center of the top surface of the top lid. Dose location 4 (mid) is situated directly above the vertical section of the outlet vent. Dose location 4 (outer) is extended along the top plane of the top lid, located radially above the overpack outer diameter.
- The “Fuel Gammas” category includes gammas from the spent fuel and  $^{60}\text{Co}$  from the spacer grids.

Table 5.1.7

DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK  
FOR NORMAL CONDITIONS  
MPC-37  
BURNUP AND COOLING TIME  
45,000 MWD/MTU AND 4.5-YEAR COOLING

<b>Dose Point Location</b>	<b>Fuel Gammas (mrem/hr)</b>	<b>(n,<math>\gamma</math>) Gammas (mrem/hr)</b>	<b><math>^{60}\text{Co}</math> Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>	<b>Totals with BPRAs (mrem/hr)</b>
1	57	1	4	1	62	62
2	75	1	1	1	77	78
3	6	<1	5	<1	13	15
4 (center)	0.6	0.3	1.0	0.2	2.1	2.7

## Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- The “Fuel Gammas” category includes gammas from the spent fuel,  $^{60}\text{Co}$  from the spacer grids, and  $^{60}\text{Co}$  from the BPRAs in the active fuel region.



Table 5.1.8

DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK  
FOR NORMAL CONDITIONS  
MPC-89  
BURNUP AND COOLING TIME  
45,000 MWD/MTU AND 5-YEAR COOLING

<b>Dose Point Location</b>	<b>Fuel Gammas (mrem/hr)</b>	<b>(n,<math>\gamma</math>) Gammas (mrem/hr)</b>	<b><math>^{60}\text{Co}</math> Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>
1	38	<1	7	<1	47
2	47	<1	<1	<1	50
3	3	<1	5	<1	10
4 (center)	0.2	0.2	1	0.1	2

## Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- The “Fuel Gammas” category includes gammas from the spent fuel and  $^{60}\text{Co}$  from the spacer girds.

Table 5.1.9

DOSE FROM HI-TRAC VW WITH MPC-37  
FOR ACCIDENT CONDITIONS  
AT 100 METERS  
65,000 MWD/MTU AND 8-YEAR COOLING

<b>Dose Point Location</b>	<b>Dose Rate (rem/hr)</b>	<b>Accident Duration (days)</b>	<b>Total Dose (rem)</b>	<b>Regulatory Limit (rem)</b>	<b>Time to Reach Regulatory Limit (days)</b>
2 (Accident Condition)	2.3E-3	30	1.66	5	90

## Notes:

- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer where appropriate.
- Dose rates used to evaluated “ Total Dose (rem)” are from Table 5.1.4
- Regulatory Limit is from 10CFR72.106.

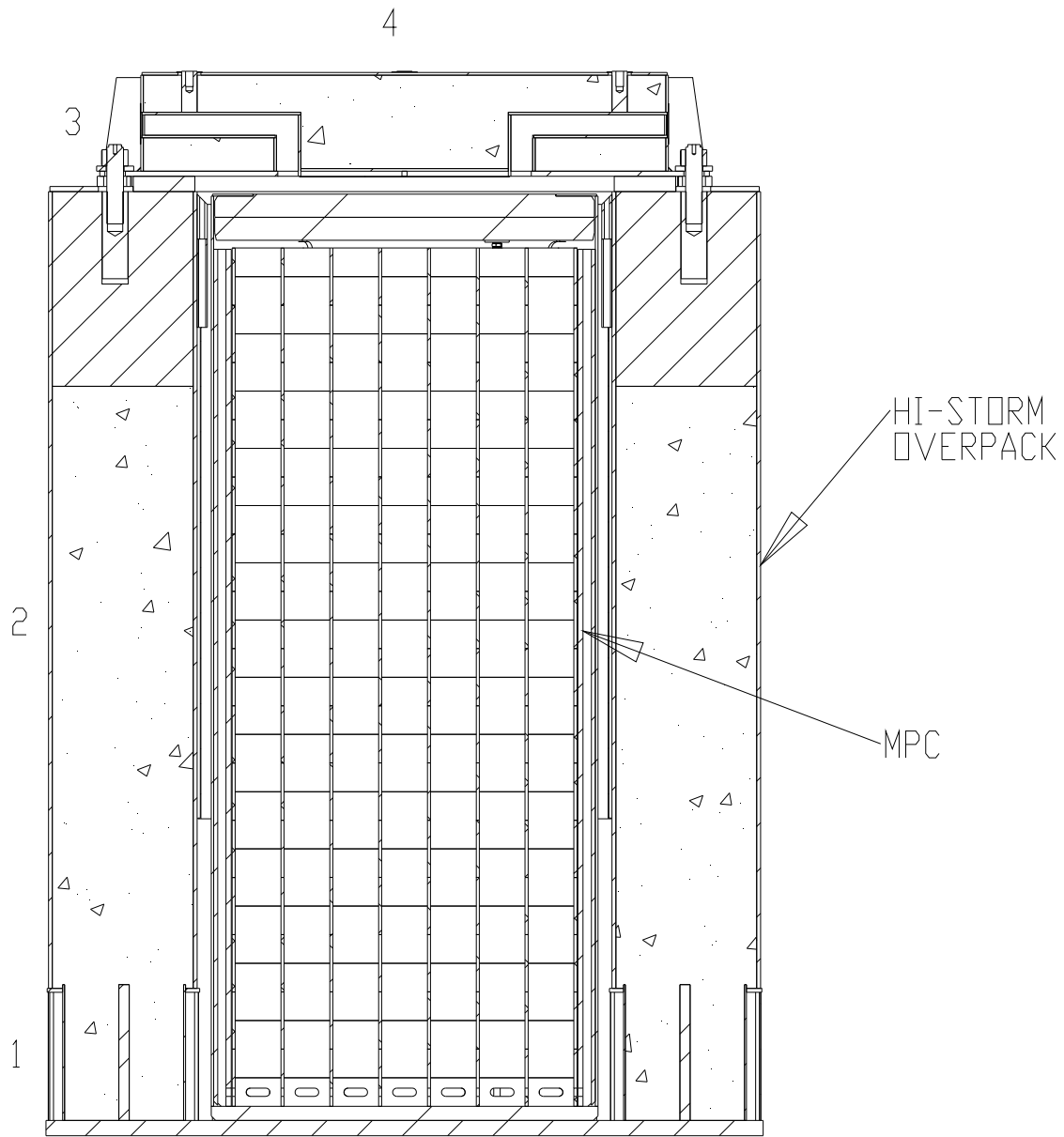


Figure 5.1.1

### CROSS SECTION ELEVATION VIEW OF HI-STORM FW OVERPACK WITH DOSE POINT LOCATIONS

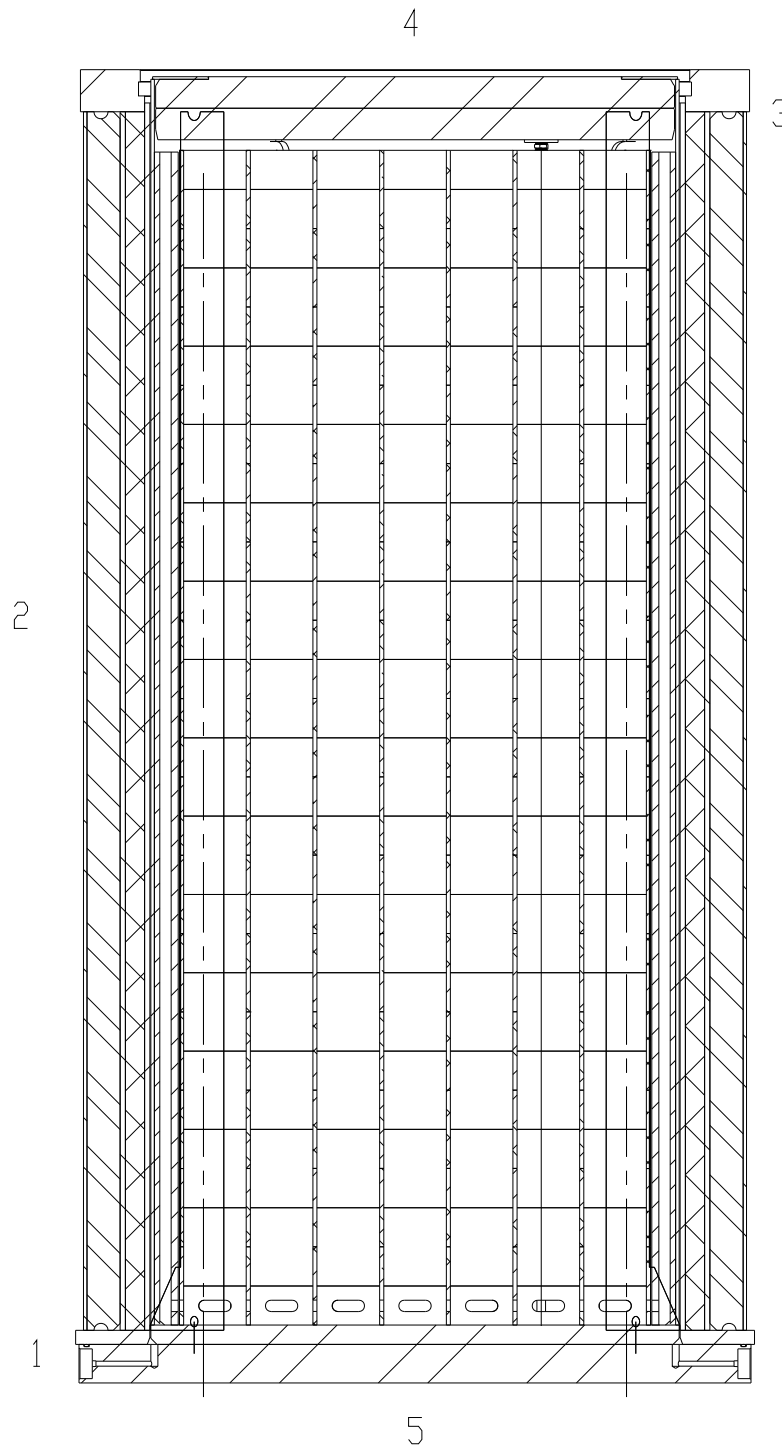


Figure 5.1.2  
CROSS SECTION ELEVATION VIEW OF HI-TRAC VW TRANSFER CASK WITH DOSE  
POINT LOCATIONS

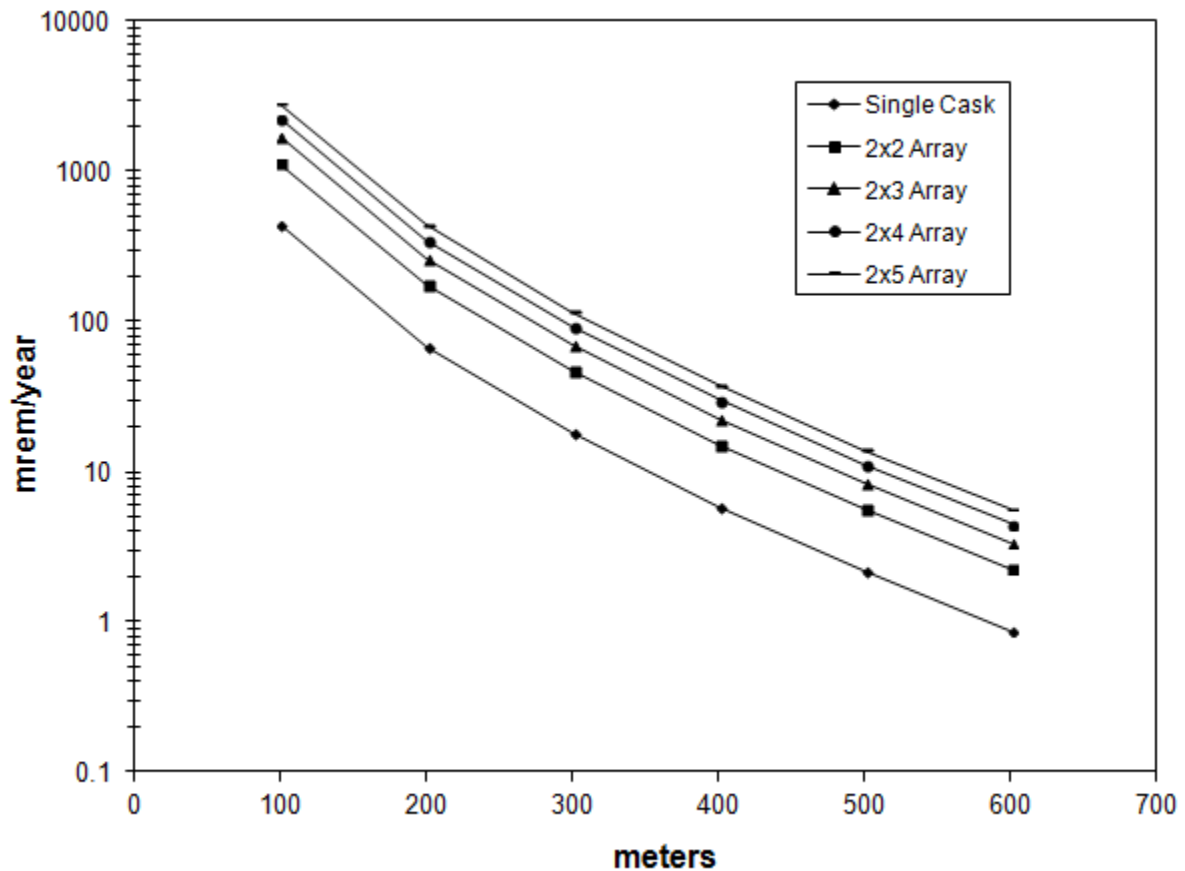


Figure 5.1.3

ANNUAL DOSE VERSUS DISTANCE FOR VARIOUS CONFIGURATIONS OF THE MPC-37 FOR 45,000 MWD/MTU AND 4.5 YEAR COOLING (8760 HOUR 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 5 system [5.1.2, 5.1.3]. SAS2H has been extensively compared to experimental isotopic validations and decay heat measurements. References [5.2.8] through [5.2.12] and [5.2.15] present isotopic comparisons for PWR and BWR fuels for burnups ranging to 47 GWD/MTU and reference [5.2.13] presents results for BWR measurements to a burnup of 57 GWD/MTU. A comparison of calculated and measured decays heats is presented in reference [5.2.14]. All of these studies indicate good agreement between SAS2H and measured data. Additional comparisons of calculated values and measured data are being performed by various institutions for high burnup PWR and BWR fuel. These new results, when published, are expected to further confirm the validity of SAS2H for the analysis of PWR and BWR fuel.

Sample input files for SAS2H and ORIGEN-S are provided in Appendix 5.A. 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  $^{60}\text{Co}$  activity of the stainless steel structural material in the fuel element above and below the active fuel region. The third source is from  $(n,\gamma)$  reactions described below.

A description of the design basis fuel for the source term calculations is provided in Table 5.2.1. Subsection 5.2.5 discusses, in detail, the determination of the design basis fuel assemblies.

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 Table 5.2.1 resulted in conservative source term calculations.

### 5.2.1 Gamma Source

Tables 5.2.2 through 5.2.5 provide the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design basis zircaloy clad fuel at the burnups and cooling times used for normal and accident conditions.

Previous analyses were performed for the HI-STORM 100 system to determine the dose contribution from gammas as a function of energy [5.2.17]. The results of these analyses have revealed that, due to the magnitude of the gamma source at lower energies, photons with energies as low as 0.45 MeV must be included in the shielding analysis, but photons with energies below 0.45 MeV are too weak to penetrate the HI-STORM overpack or HI-TRAC. The effect of gammas with energies above 3.0 MeV, on the other hand, was found to be insignificant. This is due to the fact that the source of gammas in this range (i.e., above 3.0 MeV) is extremely low. Therefore, all photons with energies in the range of 0.45 to 3.0 MeV are included in the shielding calculations.

The primary source of activity in the non-fuel regions of an assembly arises from the activation of  $^{59}\text{Co}$  to  $^{60}\text{Co}$ . The primary source of  $^{59}\text{Co}$  in a fuel assembly is impurities in the steel structural material above and below the fuel. The zircaloy in these regions is neglected since it does not have a significant  $^{59}\text{Co}$  impurity level. Reference [5.2.2] indicates that the impurity level in steel is 800 ppm or 0.8 gm/kg. Therefore, inconel and stainless steel in the non-fuel regions are both assumed to have the same 0.8 gm/kg impurity level.

Some of the PWR fuel assembly designs (B&W and WE 15x15) utilized inconel in-core grid spacers while other PWR fuel designs use zircaloy in-core grid spacers. In the mid 1980s, the fuel assembly designs using inconel in-core grid spacers were altered to use zircaloy in-core grid spacers. Since both designs may be loaded into the HI-STORM FW system, the gamma source for the PWR zircaloy clad fuel assembly includes the activation of the in-core grid spacers. Although BWR assembly grid spacers are zircaloy, some assembly designs have inconel springs in conjunction with the grid spacers. The gamma source for the BWR zircaloy clad fuel assembly includes the activation of these springs associated with the grid spacers.

The non-fuel data listed in Table 5.2.1 were taken from References [5.2.2], [5.2.4], and [5.2.5]. As stated above, a Cobalt-59 impurity level of 0.8 gm/kg was used for both inconel and stainless steel. Therefore, there is little distinction between stainless steel and inconel in the source term generation and since the shielding characteristics are similar, stainless steel was used in the MCNP calculations instead of inconel. The BWR masses for an 8x8 fuel assembly were used. These masses are also appropriate for the 10x10 assembly since the masses of the non-fuel hardware from a 10x10 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.

The masses in Table 5.2.1 were used to calculate a  $^{59}\text{Co}$  impurity level in the fuel assembly material. The grams of impurity were then used in ORIGEN-S to calculate a  $^{60}\text{Co}$  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  $^{60}\text{Co}$  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 5.2.6. These scaling factors were taken from Reference [5.2.3].

Tables 5.2.7 through 5.2.10 provide the  $^{60}\text{Co}$  activity utilized in the shielding calculations for normal and accident conditions for the non-fuel regions of the assemblies in the MPC-37 and the MPC-89.

In addition to the two sources already mentioned, a third source arises from (n, $\gamma$ ) 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 of 3.2 and 3.6 wt% were chosen for the BWR and PWR design basis fuel assemblies under normal conditions, respectively. For the accident conditions, a fuel enrichment of 4.8 wt% was chosen to accommodate the higher burnups of the selected source terms (see Table 5.0.2) in accordance with Table 5.2.24 of reference [5.2.17].

The neutron source calculated for the design basis fuel assemblies for the MPCs and the design basis fuel are listed in Tables 5.2.11 through 5.2.14 in neutrons/s for the selected burnup and cooling times used in the shielding evaluations for normal and accident conditions. The neutron spectrum is generated in ORIGEN-S.

## 5.2.3 Non-Fuel Hardware

Burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), and axial power shaping rods (APSRs) are permitted for storage in the HI-STORM FW system as an integral part of a PWR fuel assembly. BPRAs and TPDs may be stored in any fuel location while CRAs and APSRs are restricted as specified in Subsection 2.1.

### 5.2.3.1 BPRAs and TPDs

Burnable poison rod assemblies (BPRA) (including wet annular burnable absorbers) and thimble plug devices (TPD) (including orifice rod assemblies, guide tube plugs, and water displacement guide tube plugs) 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 from that of a fuel assembly. Vibration suppressor inserts are considered to be in the same category as BPRAs for the purposes of the analysis in this chapter since these devices have the same configuration (long non-absorbing thimbles which extend into the active fuel region) as a BPRA without the burnable poison.

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 typically do not extend into the active fuel region. 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 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 W 17x17 fuel assembly. The mass of material in the regions above the active fuel zone was scaled by the appropriate scaling factors listed in Table 5.2.6 in order to account for the reduced flux levels above the fuel assembly. The total curies of cobalt were calculated for the TPDs and BPRAs as a function of burnup and cooling time.

Since the HI-STORM FW cask system is designed to store many varieties of PWR fuel, a representative TPD and BPRA had to be determined for the purposes of the analysis. This was accomplished in the HI-STORM 100 FSAR [5.2.17] 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 TPD was determined to be the Westinghouse 17x17 guide tube plug and the BPRA was actually determined by combining the higher masses of the Westinghouse 17x17 and 15x15 BPRAs into a single hypothetical BPRA. The masses of these devices are listed in Table 5.2.15.

Table 5.2.16 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). A burnup and cooling time, separate from the fuel assemblies, is used for BPRAs and TPDs. Table 2.1.25 of the HI-STORM 100 [5.2.17] lists the allowable burnups and cooling times for non-fuel hardware that corresponds to the BPRA. These burnup and cooling times assure that the Co-60 activity remains below the levels specified above. For specific site boundary evaluations, these levels/values can be used if they are bounding. Alternatively, more realistic values can be used.

The HI-STORM 100 [5.2.17] presents dose rates for both BPRAs and TPDs. The results indicate that BPRAs are bounding, therefore all dose rates in this chapter will contain a BPRA in every PWR fuel location. However, Section 5.4 also contains a quantitative dose rates comparison from BPRAs and TPDs to validate this approach. Subsection 5.4.4 discusses the increase in the cask dose rates due to the insertion of BPRAs into fuel assemblies.

### **5.2.3.2 CRAs and APSRs**

Control rod assemblies (CRAs) (including control element assemblies and rod cluster control assemblies) and axial power shaping rod assemblies (APSRs) are an integral portion of a PWR fuel assembly. These devices are utilized for many years (upwards of 20 years) prior to discharge into the spent fuel pool. The manner in which the CRAs are utilized vary from plant to plant.

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Some utilities maintain the CRAs fully withdrawn during normal operation while others may operate with a bank of rods partially inserted (approximately 10%) during normal operation. Even when fully withdrawn, the ends of the CRAs are present in the upper portion of the fuel assembly since they are never fully removed from the fuel assembly during operation. The result of the different operating styles is a variation in the source term for the CRAs. In all cases, however, only the lower portion of the CRAs will be significantly activated. Therefore, when the CRAs are stored with the PWR fuel assembly, the activated portion of the CRAs will be in the lower portion of the cask. CRAs are fabricated of various materials. The cladding is typically stainless steel, although inconel has been used. The absorber can be a single material or a combination of materials. AgInCd is possibly the most common absorber although B<sub>4</sub>C in aluminum is used, and hafnium has also been used. AgInCd produces a noticeable source term in the 0.3-1.0 MeV range due to the activation of Ag. The source term from the other absorbers is negligible, therefore the AgInCd CRAs are the bounding CRAs.

APSRs are used to flatten the power distribution during normal operation and as a result these devices achieve a considerably higher activation than CRAs. There are two types of B&W stainless steel clad APSRs: gray and black. According to reference [5.2.5], the black APSRs have 36 inches of AgInCd as the absorber while the gray ones use 63 inches of inconel as the absorber. Because of the cobalt-60 source from the activation of inconel, the gray APSRs produce a higher source term than the black APSRs and therefore are the bounding APSR.

Since the level of activation of CRAs and APSRs can vary, the quantity that can be stored in an MPC is being limited. These devices are required to be stored in the locations as outlined in Subsection 2.1.

Subsection 5.4.4 discusses the effect on dose rate of the insertion of APSRs or CRAs into fuel assemblies.

#### 5.2.4 Choice of Design Basis Assembly

The Westinghouse 17x17 and GE 10x10 assemblies were selected as design basis assemblies since they are widely used throughout the industry. Site specific shielding evaluations should verify that those assemblies and assembly parameters are appropriate for the site-specific analyses.

#### 5.2.5 Decay Heat Loads and Allowable Burnup and Cooling Times

Subsection 2.1 describes the MPC maximum decay heat limits per assembly. The allowable burnup and cooling time limits are derived based on the allowable decay heat limits.

#### 5.2.6 Fuel Assembly Neutron Sources

Neutron source assemblies (NSAs) are used in reactors for startup. There are different types of neutron sources (e.g. californium, americium-beryllium, plutonium-beryllium, polonium-

beryllium, antimony-beryllium). These neutron sources are typically inserted into the water rod of a fuel assembly and are usually removable.

During in-core operations, the stainless steel and inconel portions of the NSAs become activated, producing a significant amount of Co-60. A detailed discussion about NSAs is provided in reference [5.2.17], where it is concluded that activation from NSAs are bounded by activation from BPRAs.

For ease of implementation in the CoC, the restriction concerning the number of NSAs is being applied to all types of NSAs. In addition, conservatively NSAs are required to be stored in the inner region of the MPC basket as specified in Subsection 2.1. Further limitations allow for only one NSA to be stored in the MPC-37 (see Table 2.1.1).

Table 5.2.1		
DESCRIPTION OF DESIGN BASIS CLAD FUEL		
	PWR	BWR
Assembly type/class	WE 17×17	GE 10×10
Active fuel length (in.)	144	144
No. of fuel rods	264	92
Rod pitch (in.)	0.496	0.51
Cladding material	Zircaloy-4	Zircaloy-2
Rod diameter (in.)	0.374	0.404
Cladding thickness (in.)	0.0225	0.026
Pellet diameter (in.)	0.3232	0.345
Pellet material	UO <sub>2</sub>	UO <sub>2</sub>
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.522 (96% of theoretical)
Enrichment (w/o <sup>235</sup> U)	3.6	3.2
Specific power (MW/MTU)	43.48	30
Weight of UO <sub>2</sub> (kg) <sup>††</sup>	532.150	213.531
Weight of U (kg) <sup>††</sup>	469.144	188.249
No. of Water Rods/ Guide Tubes	25	2
Water Rod/ Guide Tube O.D. (in.)	0.474	0.98
Water Rod/ Guide Tube Thickness (in.)	0.016	0.03

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<sup>††</sup> Derived from parameters in this table.

Table 5.2.1 (continued)		
DESCRIPTION OF DESIGN BASIS FUEL		
	<b>PWR</b>	<b>BWR</b>
Lower End Fitting (kg)	5.9 (steel)	4.8 (steel)
Gas Plenum Springs (kg)	1.150 (steel)	1.1 (steel)
Gas Plenum Spacer (kg)	0.793 (inconel) 0.841 (steel)	N/A
Expansion Springs (kg)	N/A	0.4 (steel)
Upper End Fitting (kg)	6.89 (steel) 0.96 (inconel)	2.0 (steel)
Handle (kg)	N/A	0.5 (steel)
Incore Grid Spacers (kg)	4.9 (inconel)	0.33 (inconel springs)

Table 5.2.2			
CALCULATED MPC-37 PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS			
Lower Energy	Upper Energy	45,000 MWD/MTU 4.5-Year Cooling	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	2.11E+15	3.68E+15
0.7	1.0	7.67E+14	9.02E+14
1.0	1.5	1.74E+14	1.39E+14
1.5	2.0	1.45E+13	8.30E+12
2.0	2.5	1.01E+13	4.47E+12
2.5	3.0	4.05E+11	1.47E+11
Total		3.08E+15	4.73E+15

Table 5.2.3			
CALCULATED MPC-37 PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS			
Lower Energy	Upper Energy	65,000 MWD/MTU 8-Year Cooling	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	2.05E+15	3.56E+15
0.7	1.0	4.16E+14	4.89E+14
1.0	1.5	1.30E+14	1.04E+14
1.5	2.0	8.66E+12	4.95E+12
2.0	2.5	6.46E+11	2.87E+11
2.5	3.0	4.49E+10	1.63E+10
Total		2.60E+15	4.16E+15

Table 5.2.4			
CALCULATED MPC-89 BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS			
Lower Energy	Upper Energy	45,000 MWD/MTU 5-Year Cooling	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	7.52E+14	1.31E+15
0.7	1.0	2.40E+14	2.82E+14
1.0	1.5	5.53E+13	4.42E+13
1.5	2.0	4.15E+12	2.37E+12
2.0	2.5	2.02E+12	8.97E+11
2.5	3.0	9.74E+10	3.54E+10
Total		1.05E+15	2.04E+15

Table 5.2.5			
CALCULATED MPC-89 BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS			
Lower Energy	Upper Energy	65,000 MWD/MTU 10-Year Cooling	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	6.98E+14	1.21E+15
0.7	1.0	8.37E+13	9.85E+13
1.0	1.5	3.50E+13	2.80E+13
1.5	2.0	2.52E+12	1.44E+12
2.0	2.5	4.49E+10	2.00E+10
2.5	3.0	3.90E+09	1.42E+09
Total		8.19E+14	1.34E+15

Table 5.2.6

SCALING FACTORS USED IN CALCULATING THE  $^{60}\text{Co}$  SOURCE

<b>Region</b>	<b>PWR</b>	<b>BWR</b>
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
Incore Grid Spacer	1.0	1.0
Lower End Fitting	0.2	0.15



Table 5.2.7

CALCULATED MPC-37  $^{60}\text{Co}$  SOURCE PER ASSEMBLY FOR DESIGN BASIS  
FUEL AT DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS

<b>Location</b>	<b>45,000 MWD/MTU and 4.5-Year Cooling (curies)</b>
Lower End Fitting	86.02
Gas Plenum Springs	16.77
Gas Plenum Spacer	11.91
Expansion Springs	NA
Incore Grid Spacers	357.19
Upper End Fitting	57.22
Handle	NA

Table 5.2.8

CALCULATED MPC-37  $^{60}\text{Co}$  SOURCE PER ASSEMBLY FOR DESIGN BASIS  
FUEL AT BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS

<b>Location</b>	<b>65,000 MWD/MTU and 8-Year Cooling (curies)</b>
Lower End Fitting	64.89
Gas Plenum Springs	12.65
Gas Plenum Spacer	8.99
Expansion Springs	NA
Incore Grid Spacers	269.46
Upper End Fitting	43.17
Handle	NA

Table 5.2.9

CALCULATED MPC-89  $^{60}\text{Co}$  SOURCE PER ASSEMBLY FOR DESIGN BASIS  
FUEL AT DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS

<b>Location</b>	<b>45,000 MWD/MTU and 5-Year Cooling (curies)</b>
Lower End Fitting	158.66
Gas Plenum Springs	48.48
Gas Plenum Spacer	N/A
Expansion Springs	8.81
Grid Spacer Springs	72.72
Upper End Fitting	44.07
Handle	5.51

Table 5.2.10

CALCULATED MPC-89  $^{60}\text{Co}$  SOURCE PER ASSEMBLY FOR DESIGN BASIS  
FUEL AT BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS

<b>Location</b>	<b>65,000 MWD/MTU and 10-Year Cooling (curies)</b>
Lower End Fitting	90.17
Gas Plenum Springs	27.55
Gas Plenum Spacer	N/A
Expansion Springs	5.01
Grid Spacer Springs	41.33
Upper End Fitting	25.05
Handle	3.13

Table 5.2.11		
CALCULATED MPC-37 PWR NEUTRON SOURCE PER ASSEMBLY FOR 45,000 MWD/MTU BURNUP AND 4.5 YEAR COOLING		
Lower Energy (MeV)	Upper Energy (MeV)	45,000 MWD/MTU 4.5-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	3.05E+07
4.0e-01	9.0e-01	6.64E+07
9.0e-01	1.4	6.63E+07
1.4	1.85	5.30E+07
1.85	3.0	9.88E+07
3.0	6.43	8.97E+07
6.43	20.0	8.56E+06
Totals		4.13E+08

Table 5.2.12		
CALCULATED MPC-37 PWR NEUTRON SOURCE PER ASSEMBLY FOR 65,000 MWD/MTU BURNUP AND 8 YEAR COOLING		
Lower Energy (MeV)	Upper Energy (MeV)	65,000 MWD/MTU 8-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	6.80E+07
4.0e-01	9.0e-01	1.48E+08
9.0e-01	1.4	1.47E+08
1.4	1.85	1.17E+08
1.85	3.0	2.18E+08
3.0	6.43	1.98E+08
6.43	20.0	1.89E+07
Totals		9.16E+08

Table 5.2.13		
CALCULATED MPC-89 BWR NEUTRON SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL FOR 45,000 MWD/MTU BURNUP AND 5 YEAR COOLING		
Lower Energy (MeV)	Upper Energy (MeV)	45,000 MWD/MTU 5-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	1.37E+07
4.0e-01	9.0e-01	2.99E+07
9.0e-01	1.4	2.99E+07
1.4	1.85	2.38E+07
1.85	3.0	4.44E+07
3.0	6.43	4.03E+07
6.43	20.0	3.86E+06
Totals		1.86E+08

Table 5.2.14		
CALCULATED MPC-89 BWR NEUTRON SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL FOR 65,000 MWD/MTU BURNUP AND 10 YEAR COOLING		
Lower Energy (MeV)	Upper Energy (MeV)	65,000 MWD/MTU 10-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	2.40E+07
4.0e-01	9.0e-01	5.22E+07
9.0e-01	1.4	5.20E+07
1.4	1.85	4.15E+07
1.85	3.0	7.71E+07
3.0	6.43	7.00E+07
6.43	20.0	6.68E+06
Totals		3.24E+08

Table 5.2.15 DESCRIPTION OF DESIGN BASIS BURNABLE POISON ROD ASSEMBLY AND THIMBLE PLUG DEVICE		
<b>Region</b>	<b>BPRA</b>	<b>TPD</b>
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 5.2.16 DESIGN BASIS COBALT-60 ACTIVITIES FOR BURNABLE POISON ROD ASSEMBLIES AND THIMBLE PLUG DEVICES		
<b>Region</b>	<b>BPRA</b>	<b>TPD</b>
Upper End Fitting (curies Co-60)	32.7	25.21
Gas Plenum Spacer (curies Co-60)	5.0	9.04
Gas Plenum Springs (curies Co-60)	8.9	15.75
In-core (curies Co-60)	848.4	N/A

## 5.3 MODEL SPECIFICATIONS

The shielding analysis of the HI-STORM FW system was performed with MCNP5 [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-STORM FW system, including the HI-TRAC transfer casks, in the shielding analysis. A sample input file for MCNP is provided in Appendix 5.A.

As discussed in Subsection 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 conditions. Subsection 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 (water) in the HI-TRAC. Therefore, the MCNP model of the normal HI-TRAC condition has the neutron shield in place while the accident condition replaces the neutron shield with void. Subsection 5.1.2 also mentioned that there is no credible accident scenario that would impact the HI-STORM shielding analysis. Therefore, models and results for the normal and accident conditions are identical for the HI-STORM overpack.

### 5.3.1 Description of the Radial and Axial Shielding Configuration

Chapter 1 provides the drawings that describe the HI-STORM FW system, including the HI-TRAC transfer cask. These drawings, using nominal dimensions, were used to create the MCNP models used in the radiation transport calculations. Modeling deviations from these drawings are discussed below. Figures 5.3.1 and 5.3.2, as well as Figures 5.3.12 and 5.3.13, show cross sectional views of the HI-STORM FW overpack, MPCs, and basket cells as they are modeled in MCNP. Figures 5.3.1 and 5.3.2 were created in VISED and are drawn to scale. The inlet and outlet vents were modeled explicitly, therefore, streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and at 1 meter. Figures 5.3.3 and 5.3.4 show a cross sectional view of the HI-TRAC VW with the MPC-37 and MPC-89, respectively, as it was modeled in MCNP. These figures were created in VISED and are drawn to scale.

Figure 5.3.5 shows a cross sectional view of the HI-STORM FW overpack with the as-modeled thickness of the various materials.

Figure 5.3.6 shows the axial representation of the HI-STORM FW overpack with the various as-modeled dimensions indicated.

Figure 5.3.7 shows axial cross-sectional views of the HI-TRAC VW transfer casks with the as-modeled dimensions and materials specified. Figures 5.3.8 and 5.3.9 shows fully labeled radial cross-sectional view of the HI-TRAC VW transfer casks and each of the MPCs.

Calculations were performed for the HI-STORM 100 [5.2.17] to determine the acceptability of homogenizing the fuel assembly versus explicit modeling. Based on these calculations it was concluded that it is acceptable to homogenize the fuel assembly without loss of accuracy. The width of the PWR and BWR homogenized fuel assembly is equal to 17 times the pitch and 10 times the pitch, respectively. Homogenization results in a noticeable decrease in run time.

Several conservative approximations were made in modeling the MPC. The conservative approximations are listed below.

1. The fuel shims are not modeled because they are not needed on all fuel assembly types. However, most PWR fuel assemblies will have fuel shims. The fuel shim length for the design basis fuel assembly type determines the positioning of the fuel assembly for the shielding analysis. This is conservative since it removes steel that would provide a small amount of additional shielding.
2. The MPC basket supports are not modeled. This is conservative since it removes material that would provide a small increase in shielding.

The zircaloy flow channels are included in the modeling of the BWR assemblies. The expected impact of this assumption on the dose rates is insignificant. Additionally, site specific analysis should consider site specific fuel characteristics as applicable.

#### **5.3.1.1 Fuel Configuration**

As described earlier, 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 5.3.10 and 5.3.11 graphically depict the location of the PWR and BWR fuel assemblies within the HI-STORM FW system. The axial locations of the basket, inlet vents, and outlet vents are shown in these figures.

#### **5.3.1.2 Streaming Considerations**

The MCNP model of the HI-STORM overpack completely describes the inlet and outlet vents, thereby properly accounting for their streaming effect. Further, the top lid is properly modeled with its reduced diameter, which accounts for higher localized dose rates on the top surface of the HI-STORM.

The MCNP model of the HI-TRAC transfer cask accounts for the fins through the HI-TRAC water jacket, as discussed in Subsection 5.4.1, as well as the open annulus.



### 5.3.2 Regional Densities

Composition and densities of the various materials used in the HI-STORM FW system and HI-TRAC shielding analyses are given in Table 5.3.2. All of the materials and their actual geometries are represented in the MCNP model.

The concrete density shown in Table 5.3.2 is the minimum concrete density analyzed in this chapter. The HI-STORM FW overpacks are designed in such a way that the concrete density in the body of the overpack can be increased to approximately  $3.2 \text{ g/cm}^3$  (200 lb/cu-ft). Increasing the density beyond the value in Table 5.3.2 would result in a significant reduction in the dose rates. This may be beneficial based on on-site and off-site ALARA considerations.

The water density inside the MPC corresponds to the maximum allowable water temperature within the MPC. The water density in the water jacket corresponds to the maximum allowable temperature at the maximum allowable pressure. As mentioned, the HI-TRAC transfer cask may be equipped with a water jacket to provide radial neutron shielding. Demineralized water (borated water) will be utilized in the water jacket. To ensure operability for low temperature conditions, ethylene glycol (25% in solution) may be added to reduce the freezing point for low temperature operations. Calculations were performed for the HI-STORM 100 system [5.2.17] to determine the effect of the ethylene glycol on the shielding effectiveness of the radial neutron shield. Based on these calculations, it was concluded that the addition of ethylene glycol (25% in solution) does not reduce the shielding effectiveness of the radial neutron shield.

Subsections 4.4 and 4.5 demonstrate that all materials used in the HI-STORM and HI-TRAC 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 Subsection 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 (water) in the HI-TRAC 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 liquid neutron shield was replaced by void.

Table 5.3.1

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE FUEL ASSEMBLIES<sup>†</sup>

Region	Start (in.)	Finish (in.)	Length (in.)	Actual Material	Modeled Material
<b>PWR</b>					
Lower End Fitting	0.0	2.738	2.738	SS304	SS304
Space	2.738	3.738	1.0	zircaloy	void
Fuel	3.738	147.738	144.0	fuel & zircaloy	fuel & zircaloy
Gas Plenum Springs	147.738	151.916	4.178	SS304 & inconel	SS304
Gas Plenum Spacer	151.916	156.095	4.179	SS304 & inconel	SS304
Upper End Fitting	156.095	159.765	3.670	SS304 & inconel	SS304
<b>BWR</b>					
Lower End Fitting	0.0	7.385	7.385	SS304	SS304
Fuel	7.385	151.385	144.0	fuel & zircaloy	fuel & zircaloy
Space	151.385	157.385	6.0	zircaloy	void
Gas Plenum Springs	157.385	166.865	9.48	SS304 & zircaloy	SS304
Expansion Springs	166.865	168.215	1.35	SS304	SS304
Upper End Fitting	168.215	171.555	3.34	SS304	SS304
Handle	171.555	176	4.445	SS304	SS304

<sup>†</sup>

All dimensions start at the bottom of the fuel assembly. The length of the fuel shims must be added to the distances to determine the distance from the top of the MPC baseplate.

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Table 5.3.2			
COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM			
Component	Density (g/cm <sup>3</sup> )	Elements	Mass Fraction (%)
Metamic-HT <sup>†</sup>	2.61 (9% B <sub>4</sub> C)	<b>Withheld in Accordance with 10 CFR 2.390</b>	
SS304	7.94	Cr	19
		Mn	2
		Fe	69.5
		Ni	9.5
Carbon Steel	7.82	C	1.0
		Fe	99.0
Zircaloy	6.55	Zr	98.24
		Sn	1.45
		Fe	0.21
		Cr	0.10

<sup>†</sup> All B-10 loadings in the Metamic compositions are conservatively lower than the values defined in the Bill of Materials.

Table 5.3.2 (continued)			
COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM			
Component	Density (g/cm <sup>3</sup> )	Elements	Mass Fraction (%)
BWR Fuel Region Mixture	4.781 (5.0 wt% U-235)	<sup>235</sup> U	3.207
		<sup>238</sup> U	60.935
		O	8.623
		Zr	26.752
		N	0.014
		Cr	0.027
		Fe	0.034
		Sn	0.409
PWR Fuel Region Mixture	3.769 (5.0 wt% U-235)	<sup>235</sup> U	3.709
		<sup>238</sup> U	70.474
		O	9.972
		Zr	15.565
		Cr	0.016
		Fe	0.033
		Sn	0.230

Table 5.3.2 (continued)			
COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM			
Component	Density (g/cm <sup>3</sup> )	Elements	Mass Fraction (%)
Lower End Fitting (PWR)	1.849	SS304	100
Gas Plenum Springs (PWR)	0.23626	SS304	100
Gas Plenum Spacer (PWR)	0.33559	SS304	100
Upper End Fitting (PWR)	1.8359	SS304	100
Lower End Fitting (BWR)	1.5249	SS304	100
Gas Plenum Springs (BWR)	0.27223	SS304	100
Expansion Springs (BWR)	0.69514	SS304	100
Upper End Fitting (BWR)	1.4049	SS304	100
Handle (BWR)	0.26391	SS304	100
Lead	11.3	Pb	99.9
		Cu	0.08
		Ag	0.02
Water	0.919 (water jacket)	H	11.2
	0.958 (inside MPC)	O	88.8

Table 5.3.2 (continued)			
COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM			
Component	Density (g/cm <sup>3</sup> )	Elements	Mass Fraction (%)
Water w/ 2000 ppm	0.958	B-10	0.036
		B-11	0.164
		H	11.17
		O	88.63
Concrete	2.4	H	1.0
		O	53.2
		Si	33.7
		Al	3.4
		Na	2.9
		Ca	4.4
		Fe	1.4

**Withheld in Accordance with 10 CFR 2.390**

Figure 5.3.1  
HI-STORM FW OVERPACK WITH MPC-37 CROSS SECTIONAL VIEW AS MODELED IN  
MCNP<sup>†</sup>

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<sup>†</sup> This figure is drawn to scale using VISED.

**Withheld in Accordance with 10 CFR 2.390**

Figure 5.3.2

HI-STORM FW OVERPACK WITH MPC-89 CROSS SECTIONAL VIEW AS MODELED IN MCNP<sup>†</sup>

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<sup>†</sup> This figure is drawn to scale using VISED.



**Withheld in Accordance with 10 CFR 2.390**

Figure 5.3.3

HI-TRAC VW OVERPACK WITH MPC-37 CROSS SECTIONAL VIEW AS MODELED IN  
MCNP<sup>†</sup>

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<sup>†</sup> This figure is drawn to scale using VISED.

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Figure 5.3.4

HI-TRAC VW OVERPACK WITH MPC-89 CROSS SECTIONAL VIEW AS MODELED IN  
MCNP<sup>†</sup>

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<sup>†</sup> This figure is drawn to scale using VISED.

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Figure 5.3.5  
CROSS SECTION OF HI-STORM FW OVERPACK

**Withheld in Accordance with 10 CFR 2.390**

Figure 5.3.6  
HI-STORM FW OVERPACK CROSS SECTIONAL ELEVATION VIEW

**Withheld in Accordance with 10 CFR 2.390**

Figure 5.3.7  
HI-TRAC VW TRANSFER CASK WITH POOL LID CROSS SECTIONAL ELEVATION  
VIEW (AS MODELED)

**Withheld in Accordance with 10 CFR 2.390**

Figure 5.3.8

HI-TRAC VW TRANSFER CASK CROSS SECTIONAL VIEW WITH MPC-37 (AS  
MODELED)

**Withheld in Accordance with 10 CFR 2.390**

Figure 5.3.9

HI-TRAC VW TRANSFER CASK CROSS SECTIONAL VIEW WITH MPC-89 (AS  
MODELED)

**Withheld in Accordance with 10 CFR 2.390**

Figure 5.3.10

AXIAL LOCATION OF PWR DESIGN BASIS FUEL IN THE HI-STORM FW OVERPACK



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Figure 5.3.11

AXIAL LOCATION OF BWR DESIGN BASIS FUEL IN THE HI-STORM FW OVERPACK

**Withheld in Accordance with 10 CFR 2.390**

Figure 5.3.12

CROSS SECTIONAL VIEW OF AN MPC-37 BASKET CELL AS MODELED IN MCNP

**Withheld in Accordance with 10 CFR 2.390**

Figure 5.3.13

CROSS SECTIONAL VIEW OF AN MPC-89 BASKET CELL AS MODELED IN MCNP

## 5.4 SHIELDING EVALUATION

The MCNP-5 code was used for all of the shielding analyses [5.1.1]. MCNP is a continuous energy, three-dimensional, coupled neutron-photon-electron Monte Carlo transport code. Continuous energy cross section data are represented with sufficient energy points to permit linear-linear interpolation between points. The individual cross section libraries used for each nuclide are those recommended by the MCNP manual. Cross section libraries are based on ENDF/B-V and ENDF/B-VI, except for Sn isotopes where the ENDL92 library is used, and uranium isotopes where LANL/T16 libraries are used. These are the default libraries for the MCNP code version used here [5.1.1]. 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  $^{60}\text{Co}$ ). The axial distribution of the fuel source term is described in Table 2.1.5 and Figures 2.1.3 and 2.1.4. The PWR and BWR axial burnup distributions were obtained from References [5.4.5] and [5.4.6], respectively and have previously been utilized in the HI-STORM FSAR [5.2.17]. 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  $^{60}\text{Co}$  source in the 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 2.1.5 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 2.1.5 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 doses at the various desired locations. MCNP calculates neutron or photon flux and these values 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 in Appendix 5.A. 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].

The dose rates at the various locations were calculated with MCNP using a two-step process. The first step was to calculate the dose rate for each dose location per starting particle for each neutron and gamma group in each basket region for each axial and azimuthal dose location. The

second step is to multiply the dose rate per starting particle for each energy group and basket location (i.e., tally output/quantity) by the source strength (i.e. particles/sec) in that group and sum the resulting dose rates for all groups and basket locations in each dose location. The normalization of these results and calculation of the total dose rate from neutrons, fuel gammas or Co-60 gammas is performed with the following equation.

$$T_{final} = \sum_{j=1}^M \left[ \sum_{i=1}^N \frac{T_{i,j}}{Fm_i} * F_{i,j} \right] \quad (\text{Equation 5.4.1})$$

where,

$T_{final}$  = Final dose rate (rem/h) from neutrons, fuel gammas, or Co-60

$N$  = Number of groups (neutrons, fuel gammas) or Number of axial sections (Co-60 gammas)

$M$  = Number of regions in the basket

$T_{i,j}$  = Tally quantity from particles originating in MCNP in group/section  $i$  and region  $j$  (rem/h)(particles/sec)

$F_{i,j}$  = Fuel Assembly source strength in group  $i$  and region  $j$  (particles/sec)

$Fm_i$  = Source fraction used in MCNP for group  $i$

Note that dividing by  $Fm_i$  (normalization) is necessary to account for the number of MCNP particles that actually start in group  $i$ . Also note that  $T_i$  is already multiplied by a dose conversion factor in MCNP.

The standard deviations of the various results were statistically combined to determine the standard deviation of the total dose in each dose location. The estimated variance of the total dose rate,  $S_{total}^2$ , is the sum of the estimated variances of the individual dose rates  $S_i^2$ . The estimated total dose rate, estimated variance, and relative error [5.1.1] are derived according to Equations 5.4.2 through 5.4.5.

$$R_i = \frac{\sqrt{S_i^2}}{T_i} \quad (\text{Equation 5.4.2})$$

$$S_{Total}^2 = \sum_{i=1}^n S_i^2 \quad (\text{Equation 5.4.3})$$

$$T_{Total} = \sum_{i=1}^n T_i \quad (\text{Equation 5.4.4})$$

$$R_{Total} = \frac{\sqrt{S_{Total}^2}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^n S_i^2}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^n (R_i \times T_i)^2}}{T_{Total}} \quad (\text{Equation 5.4.5})$$

where,

$i$	=	tally component index
$n$	=	total number of components
$T_{Total}$	=	total estimated tally
$T_i$	=	tally $i$ component
$S_{Total}^2$	=	total estimated variance
$S_i^2$	=	variance of the $i$ component
$R_i$	=	relative error of the $i$ component
$R_{Total}$	=	total estimated relative error

Note that the two-step approach outlined above allows the accurate consideration of the neutron and gamma source spectrum, and the location of the individual assemblies, since the tallies are calculated in MCNP as a function of the starting energy group and the assembly location, and then in the second step multiplied with the source strength in each group in each location. It is therefore equivalent to a one-step calculation where source terms are directly specified in the MCNP input files, except for the following approximations:

The first approximation is that fuel is modeled as fresh  $\text{UO}_2$  fuel (rather than spent fuel) in MCNP, with an upper bound enrichment. The second approximation is related to the axial burnup profile. The profile is modeled by assigning a source probability to each of the 10 axial sections of the active region, based on a representative axial burnup profile [5.2.17]. For fuel gammas, the probability is proportional to the burnup, since the gamma source strength changes essentially linearly with burnup. For neutrons, the probability is proportional to the burnup raised to the power of 4.2, since the neutron source strength is proportional to the burnup raised to about that power [5.4.7]. This is a standard approach that has been previously used in the licensing calculations for the HI-STAR 100 cask [5.4.8] and HI-STORM 100 system [5.2.17].

Tables 5.1.6 and 5.1.7 provide the design basis dose rates adjacent to the HI-STORM overpack during normal conditions for the MPC types in Table 1.0.1. Table 5.1.8 provides the design basis dose rates at one meter from the overpack containing the MPC-37. A detailed discussion of the normal, off-normal, and accident condition dose rates is provided in Subsections 5.1.1 and 5.1.2.

Table 5.4.2 shows the corresponding dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-37 condition with an empty water-jacket (condition in which the HI-TRAC is removed from the spent fuel pool). Table 5.4.3 shows the dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-37 condition with the water

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jacket filled with water (condition in which welding operations are performed). For the conditions involving a fully flooded MPC-37, the internal water level was 5 inches below the MPC lid. These dose rates represent the various conditions of the HI-TRAC during operations. Comparing these results to Table 5.1.1 (dry MPC-37 and HI-TRAC water jacket filled with water) indicates that the dose rates in the upper and lower portions of the HI-TRAC are significantly reduced with water in the MPC.

Table 5.4.4 shows the corresponding dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-89 condition with an empty water-jacket. Table 5.4.5 shows the dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-89 condition with the water jacket filled with water. These results demonstrate that the dose rates on contact at the top and bottom of the HI-TRAC VW are somewhat higher in the MPC-89 case than in the MPC-37 case. However, the MPC-37 produces higher dose rates than the MPC-89 at the center of the HI-TRAC, on-contact, and at locations 1 meter away from the HI-TRAC. Therefore, the MPC-37 is used for the exposure calculations in Chapter 11 of the SAR.

The calculations presented herein are using a uniform loading pattern. All MPCs, however, also offer regionalized loading patterns, as mentioned in Section 5.0 and described in Subsection 1.2. These loading patterns authorize fuel of higher decay heat (i.e., higher burnups and shorter cooling times) to be stored in certain regions of the basket. Evaluations have been performed for the HI-STORM 100 [5.2.17] where analysis of the MPC-32 and MPC-68 using the same burnup and cooling times in Region 1 and Region 2. Region 1 contains 38% of total number of assemblies for the MPC-32 and 47% for the MPC-68. The evaluations show that approximately 21% and 27% of the neutron dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32 and MPC-68, respectively. Further, approximately 1% and 2% of the photon dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32 and MPC-68, respectively. These results clearly indicate that the outer fuel assemblies shield almost the entire gamma source from the inner assemblies in the radial direction and a significant percentage of the neutron source. The conclusion from this analysis is that the total dose rate on the external radial surfaces of the cask can be greatly reduced by placing longer cooled and lower burnup fuels on the outside of the basket. Using a uniform loading pattern, rather than employing the regionalized loading scheme, in these HI-STORM FW calculations is therefore acceptable as it produces conservative dose rate values on the radial surfaces.

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 5% and the relative error for the individual dose components was typically less than 10%.

### 5.4.1 Streaming Through Radial Steel Fins

The HI-STORM FW overpack and the HI-TRAC VW cask utilize radial steel fins for structural support and cooling. The attenuation of neutrons through steel is substantially less than the attenuation of neutrons through concrete and water. Therefore, it is possible to have neutron streaming through the fins that could result in a localized dose peak. The reverse is true for photons, which would result in a localized reduction in the photon dose.

Analysis of the steel fins in the HI-TRAC has previously been performed in the HI-STORM 100 FSAR [5.2.17] and indicates that neutron streaming is noticeable at the surface of the cask. The neutron dose rate on the surface of the steel fin is somewhat higher than the circumferential average dose rate at that location. The gamma dose rate, however, is slightly lower than the circumferential average dose rate at that location. At one meter from the cask surface there is little difference between the dose rates calculated over the fins compared to the other areas of the water jackets.

These conclusions indicate that localized neutron streaming is noticeable on the surface of the transfer casks. However, at one meter from the surface the streaming has dissipated. Since most HI-TRAC operations will involve personnel moving around the transfer cask at some distance from the cask, only surface average dose rates are reported in this chapter.

### 5.4.2 Damaged Fuel Post-Accident Shielding Evaluation

The Holtec Generic PWR and BWR DFCs are designed to accommodate any PWR or BWR fuel assembly that can physically fit inside the DFC. Damaged fuel assemblies under normal conditions, for the most part, resemble intact fuel assemblies from a shielding perspective. Under accident conditions, it can not be guaranteed that the damaged fuel assembly will remain intact. As a result, the damaged fuel assembly may begin to resemble fuel debris in its possible configuration after an accident.

Since damaged fuel is identical to intact fuel from a shielding perspective no specific analysis is required for damaged fuel under normal conditions. However, a generic shielding evaluation was previously performed for the HI-STORM 100 [5.2.17] to demonstrate that fuel debris under normal or accident conditions, or damaged fuel in a post-accident configuration, will not result in a significant increase in the dose rates around the 100-ton HI-TRAC. Since the 100-ton HI-TRAC and the HI-TRAC VW are similar in design, the conclusions from the 100-ton HI-RAC evaluations are also applicable to the HI-TRAC VW.

The scenario analyzed to determine the potential change in dose rate as a result of fuel debris or a damaged fuel assembly collapse in the HI-STORM 100 [5.2.17] feature fuel debris or a damaged fuel assembly that has collapsed (which can have a higher average fuel density than an intact fuel assembly). If the damaged fuel assembly would fully or partially collapse, the fuel density in one portion of the assembly would increase and the density in the other portion of the

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assembly would decrease. The analysis consisted of modeling the fuel assemblies in the damaged fuel locations in the MPC-24 and MPC-68 with a fuel density that was twice the normal fuel density and correspondingly increasing the source rate for these locations by a factor of two. A flat axial power distribution was used which is approximately representative of the source distribution if the top half of an assembly collapsed into the bottom half of the assembly. Increasing the fuel density over the entire fuel length, rather than in the top half or bottom half of the fuel assembly, is conservative and provides the dose rate change in both the top and bottom portion of the cask.

The results for the MPC-24 and MPC-68 calculations [5.2.17] show that the potential effect on the dose rate is not very significant for the storage of damaged fuel and/or fuel debris. This conclusion is further reinforced by the fact that the majority of the significantly damaged fuel assemblies in the spent fuel inventories are older assemblies from the earlier days of nuclear plant operations. Therefore, these assemblies will have a considerably lower burnup and longer cooling times than the assemblies analyzed in this chapter. In addition, since the dose rate change is not significant for the 100-ton HI-TRAC, the dose rate change will not be significant for the HI-TRAC VW or the HI-STORM FW overpacks.

### 5.4.3 Site Boundary Evaluation

NUREG-1536 [5.2.1] states that detailed calculations need not be presented since SAR 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 characteristics would be factored into the evaluation performed by the licensee.

The methodology of calculating the dose from a single HI-STORM overpack loaded with an MPC and various arrays of loaded HI-STORMs at distances equal to and greater than 100 meters is described in the HI-STORM 100 FSAR [5.2.17]. A back row factor of 0.20 was calculated in [5.2.17], and utilized herein to calculate dose value C below, based on the results that the dose from the side of the back row of casks is approximately 16 % of the total dose.

The annual dose, assuming 100% occupancy (8760 hours), at 300 meters from a single HI-STORM FW cask is presented in Table 5.4.6 for the design basis burnup and cooling time analyzed.

The annual dose, assuming 8760 hour occupancy, at distance from an array of casks was calculated in three steps.

1. The annual dose from the radiation leaving the side of the HI-STORM FW overpack was calculated at the distance desired. Dose value = A.

2. The annual dose from the radiation leaving the top of the HI-STORM FW overpack was calculated at the distance desired. Dose value = B.
3. The annual dose from the radiation leaving the side of a HI-STORM FW overpack, when it is behind another cask, was calculated at the distance desired. The casks have an assumed 15-foot pitch. Dose value = C.

The doses calculated in the steps above are listed in Table 5.4.7. Using these values, the annual dose (at the center of the long side) from an arbitrary 2 by Z array of HI-STORM FW overpacks can easily be calculated. The following formula describes the method.

$Z$  = number of casks along long side

$$\text{Dose} = ZA + 2ZB + ZC$$

The results for various typical arrays of HI-STORM overpacks can be found in Section 5.1. While the off-site dose analyses were performed for typical arrays of casks containing design basis fuel, compliance with the requirements of 10CFR72.104(a) can only be demonstrated on a site-specific basis, as stated earlier. Therefore, a site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10CFR72.212. The site-specific evaluation will consider the site-specific characteristics (such as exposure duration and the number of casks deployed), dose from other portions of the facility and the specifics of the fuel being stored (burnup and cooling time).

#### 5.4.4 Non-Fuel Hardware

As discussed in Subsection 5.2.3, non-fuel hardware in the form of BPRAs, TPDs, CRAs, and APSRs are permitted for storage, integral with a PWR fuel assembly, in the HI-STORM FW system. Since each device occupies the same location within an assembly, only one device will be present in a given assembly. ITTRs, which are installed after core discharge and do not contain radioactive material, may also be stored in the assembly. BPRAs, TPDs and ITTRs are authorized for unrestricted storage in an MPC. The permissible locations of the CRAs and APSRs are shown in Figure 2.1.5.

Table 5.4.8 provides the dose rates at various locations on the surface and one meter from the HI-TRAC VW due to the BPRAs and TPDs for the MPC-37. The results in Table 5.4.8 show that the BPRAs essentially bound TPDs. All dose rates with NFH in this chapter therefore assume BPRA in every assembly. Note that, even for calculations without NFH, the dose from the active region conservatively contains the contribution of the BPRA. This mainly affects dose location 1 and 2, and results for these locations are therefore identical in most tables, and don't show the dose rate difference indicated in Table 5.4.8.

The analyses in this chapter that consider presence of BPRAs assume that a full-length rod with burnable poison is present in all principal locations. In reality, many BPRAs contain full-length poison rods in some locations, and thimble rodlets in others. The burnup and cooling time combinations listed in Table 2.1.25 of HI-STORM 100 FSAR [5.2.17] for BPRAs and TPDs were selected to ensure the Co-60 activity of those devices is below the value of 895 Ci (BPRA) and 50 Ci (TPD). These activities are used in the dose evaluations presented in this chapter. Apart from the total activity, the axial distribution of the material in those devices is important for the dose rates. This axial distribution is shown in Table 5.2.15 (masses) and 5.2.16 (activities). It can be observed from Table 5.2.16, while TPDs have a lower overall activity, their activity in the gas plenum region of the assembly is higher compared to that of the BPRAs. These activities were used to calculate the dose rates in Table 5.4.8. The results in this table show that the maximum dose effect for BPRAs is at the side of the cask, while the maximum dose effect of TPDs is near and on the top of the cask. Nevertheless, Table 5.4.8 demonstrates that even near and on the top of the cask, the TPD doses are bounded by the BPRA doses. It is to be noted that BPRAs with several thimble plugs may result higher dose rate near and on the top of the cask than that reported in Table 5.4.8. However, the potential local increase in dose near and on the top of the cask due the presence of several thimble plug rodlets instead of full length BPRA rods would be more than compensated by the reduction of the dose from the side of the cask at larger distances. Therefore, using BPRAs with all burnable poison rods in the analyses that demonstrate compliance with the site boundary dose limits would be bounding, and hence the burnup and cooling time combinations for BPRAs in Table 2.1.25 of the HI-STORM 100 FSAR [5.2.17] are conservative.

Two different configurations were analyzed for CRAs and three different configurations were analyzed for APSRs in the HI-STORM FSAR [5.2.17]. The dose rate due to CRAs and APSRs was explicitly calculated for dose locations around the HI-TRAC and results were provided for the different configurations of CRAs and APSRs, respectively, in the MPCs. These results indicate the dose rate on the radial surfaces of the overpack due to the storage of these devices is less than the dose rate from BPRAs (the increase in dose rate on the radial surface due to CRAs and APSRs are virtually negligible). For the surface dose rate at the bottom, the value for the CRA is comparable to or higher than the value from the BPRA. The increase in the bottom dose rates due to the presence of CRAs is on the order of 10-15% (based on bounding configuration 1 in [5.2.17]). The dose rate out the top of the overpack is essentially 0. The latter is due to the fact that CRAs and APSRs do not achieve significant activation in the upper portion of the devices due to the manner in which they are utilized during normal reactor operations. In contrast, the dose rate out the bottom of the overpack is substantial due to these devices. However, these dose rates occur in an area (below the pool lid and transfer doors) which is not normally occupied.

While the evaluations described above are based on conservative assumptions, the conclusions can vary slightly depending on the number of CRAs and their operating conditions.

#### 5.4.5 Effect of Uncertainties

The design basis calculations presented in this chapter are based on a range of conservative assumptions, but do not explicitly account for uncertainties in the methodologies, codes and input parameters, that is, it is assumed that the effect of uncertainties is small compared to the numerous conservatisms in the analyses. To show that this assumption is valid, calculations have previously been performed as “best estimate” calculations and with estimated uncertainties added [5.4.9]. In all scenarios considered (e.g., evaluation of conservatisms in modeling assumptions, uncertainties associated with MCNP as well as the depletion analysis (including input parameters), etc.), the total dose rates long with uncertainties are comparable to, or lower than, the corresponding values from the design basis calculations. This provides further confirmation that the design basis calculations are reasonable and conservative.

Table 5.4.1 FLUX-TO-DOSE CONVERSION FACTORS (FROM [5.4.1])	
<b>Gamma Energy (MeV)</b>	<b>(rem/hr)/ (photon/cm<sup>2</sup>-s)</b>
0.01	3.96E-06
0.03	5.82E-07
0.05	2.90E-07
0.07	2.58E-07
0.1	2.83E-07
0.15	3.79E-07
0.2	5.01E-07
0.25	6.31E-07
0.3	7.59E-07
0.35	8.78E-07
0.4	9.85E-07
0.45	1.08E-06
0.5	1.17E-06
0.55	1.27E-06
0.6	1.36E-06
0.65	1.44E-06
0.7	1.52E-06
0.8	1.68E-06
1.0	1.98E-06
1.4	2.51E-06
1.8	2.99E-06
2.2	3.42E-06

Table 5.4.1 (continued)	
FLUX-TO-DOSE CONVERSION FACTORS (FROM [5.4.1])	
<b>Gamma Energy (MeV)</b>	<b>(rem/hr)/ (photon/cm<sup>2</sup>-s)</b>
2.6	3.82E-06
2.8	4.01E-06
3.25	4.41E-06
3.75	4.83E-06
4.25	5.23E-06
4.75	5.60E-06
5.0	5.80E-06
5.25	6.01E-06
5.75	6.37E-06
6.25	6.74E-06
6.75	7.11E-06
7.5	7.66E-06
9.0	8.77E-06
11.0	1.03E-05
13.0	1.18E-05
15.0	1.33E-05

Table 5.4.1 (continued)		
FLUX-TO-DOSE CONVERSION FACTORS (FROM [5.4.1])		
Neutron Energy (MeV)	Quality Factor	(rem/hr) <sup>†</sup> /(n/cm <sup>2</sup> -s)
2.5E-8	2.0	3.67E-6
1.0E-7	2.0	3.67E-6
1.0E-6	2.0	4.46E-6
1.0E-5	2.0	4.54E-6
1.0E-4	2.0	4.18E-6
1.0E-3	2.0	3.76E-6
1.0E-2	2.5	3.56E-6
0.1	7.5	2.17E-5
0.5	11.0	9.26E-5
1.0	11.0	1.32E-4
2.5	9.0	1.25E-4
5.0	8.0	1.56E-4
7.0	7.0	1.47E-4
10.0	6.5	1.47E-4
14.0	7.5	2.08E-4
20.0	8.0	2.27E-4

†

Includes the Quality Factor.

Table 5.4.2

DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION  
WITH AN EMPTY NEUTRON SHIELD  
MPC-37 DESIGN BASIS ZIRCALOY CLAD FUEL AT  
45,000 MWD/MTU AND 4.5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	$^{60}\text{Co}$ Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
<b>ADJACENT TO THE HI-TRAC VW</b>						
1	711	<1	489	67	1268	1268
2	2242	2	<1	319	2563	2563
3	7	<1	128	2	139	210
4	15	<1	227	<1	244	371
5 (bottom lid)	465	<1	1802	79	2346	2346
<b>ONE METER FROM THE HI-TRAC VW</b>						
1	541	<1	61	53	657	657
2	1141	<1	5	116	1263	1264
3	191	<1	75	20	286	326
4	8	<1	127	<1	137	208
5	259	<1	985	20	1266	1266

## Notes:

- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.
- The “Fuel Gammas” category includes gammas from the spent fuel,  $^{60}\text{Co}$  from the spacer grids, and  $^{60}\text{Co}$  from the BPRAs in the active fuel region.



Table 5.4.3

**DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION  
WITH A FULL NEUTRON SHIELD  
MPC-37 DESIGN BASIS ZIRCALOY CLAD FUEL AT  
45,000 MWD/MTU AND 4.5-YEAR COOLING**

<b>Dose Point Location</b>	<b>Fuel Gammas (mrem/hr)</b>	<b>(n,<math>\gamma</math>) Gammas (mrem/hr)</b>	<b><math>^{60}\text{Co}</math> Gammas (mrem/hr)</b>	<b>Neutrons (mrem/hr)</b>	<b>Totals (mrem/hr)</b>	<b>Totals with BPRAs (mrem/hr)</b>
<b>ADJACENT TO THE HI-TRAC VW</b>						
1	433	1	301	5	740	740
2	1266	5	<1	27	1298	1298
3	2	<1	69	<1	73	111
4	15	<1	227	<1	244	371
5 (bottom lid)	465	<1	1802	80	2347	2347
<b>ONE METER FROM THE HI-TRAC VW</b>						
1	294	1	34	4	333	334
2	657	2	3	10	671	672
3	95	<1	41	1	138	160
4	8	<1	127	<1	137	208
5	259	<1	985	20	1265	1265

## Notes:

- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.
- The “Fuel Gammas” category includes gammas from the spent fuel,  $^{60}\text{Co}$  from the spacer grids, and  $^{60}\text{Co}$  from the BPRAs in the active fuel region.

Table 5.4.4					
DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH AN EMPTY NEUTRON SHIELD MPC-89 DESIGN BASIS ZIRCALOY CLAD FUEL AT 45,000 MWD/MTU AND 5-YEAR COOLING					
Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	$^{60}\text{Co}$ Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW					
1	195	<1	1513	43	1752
2	2435	3	<1	579	3018
3	<1	<1	351	2	355
4	3	<1	217	<1	222
5 (bottom lid)	40	<1	1530	5	1576
ONE METER FROM THE HI-TRAC VW					
1	387	<1	197	69	654
2	1180	<1	12	168	1361
3	118	<1	154	26	299
4	<1	<1	132	<1	135
5	19	<1	864	2	886

## Notes:

- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.
- The “Fuel Gammas” category includes gammas from the spent fuel and  $^{60}\text{Co}$  from the spacer grids.

Table 5.4.5					
DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH A FULL NEUTRON SHIELD MPC-89 DESIGN BASIS ZIRCALOY CLAD FUEL AT 45,000 MWD/MTU AND 5-YEAR COOLING					
Dose Point Location	Fuel Gammas (mrem/hr)	(n, $\gamma$ ) Gammas (mrem/hr)	$^{60}\text{Co}$ Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
<b>ADJACENT TO THE HI-TRAC VW</b>					
1	102	<1	926	2	1031
2	1373	10	<1	47	1431
3	<1	<1	189	<1	192
4	3	<1	217	<1	222
5 (bottom lid)	40	<1	1530	5	1576
<b>ONE METER FROM THE HI-TRAC VW</b>					
1	211	1	115	5	332
2	627	3	7	16	653
3	72	<1	83	1	157
4	<1	<1	132	<1	135
5	19	<1	864	2	886

## Notes:

- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.
- The “Fuel Gammas” category includes gammas from the spent fuel and  $^{60}\text{Co}$  from the spacer grids.

Table 5.4.6	
ANNUAL DOSE AT 300 METERS FROM A SINGLE HI-STORM FW OVERPACK WITH AN MPC-37 WITH DESIGN BASIS ZIRCALOY CLAD FUEL	
<b>Dose Component</b>	<b>45,000 MWD/MTU 4.5-Year Cooling (mrem/yr)</b>
Fuel gammas	15.8
<sup>60</sup> Co Gammas	2.2
Neutrons	0.2
Total	18.2

## Notes:

- Gammas generated by neutron capture are included with fuel gammas.
- The Co-60 gammas include BPRAs.
- The “Fuel Gammas” category includes gammas from the spent fuel, <sup>60</sup>Co from the spacer grids, and <sup>60</sup>Co from the BPRAs in the active fuel region.

<p style="text-align: center;">Table 5.4.7</p> <p style="text-align: center;">DOSE VALUES USED IN CALCULATING ANNUAL DOSE FROM VARIOUS HI-STORM FW ISFSI CONFIGURATIONS 45,000 MWD/MTU AND 4.5-YEAR COOLING ZIRCALOY CLAD FUEL</p>			
<b>Distance</b>	<b>A Side of Overpack (mrem/yr)</b>	<b>B Top of Overpack (mrem/yr)</b>	<b>C Side of Shielded Overpack (mrem/yr)</b>
100 meters	396.0	44.0	79.2
200 meters	61.7	6.9	12.3
300 meters	16.4	1.8	3.3
400 meters	5.3	0.6	1.1
500 meters	2.0	0.2	0.4
600 meters	0.8	0.1	0.2

## Notes:

- 8760 hour annual occupancy is assumed.
- Values are rounded to nearest integer where appropriate.

Table 5.4.8 DOSE RATES DUE TO BPRAs AND TPDs FROM THE HI-TRAC VW FOR NORMAL CONDITIONS		
Dose Point Location	BPRAs (mrem/hr)	TPDs (mrem/hr)
<b>ADJACENT TO THE HI-TRAC VW</b>		
1	159.09	0.0
2	509.04	0.0
3	192.78	165.31
4	304.15	275.53
5	137.27	0.0
<b>ONE METER FROM THE HI-TRAC VW</b>		
1	122.06	0.40
2	240.70	3.10
3	128.50	86.95
4	174.25	153.49
5	63.13	0.0

## Notes:

- Refer to Figure 5.1.2 for dose locations.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate
- Includes the BPRAs from both the active and non-active region.

## 5.5 REGULATORY COMPLIANCE

Chapters 1 and 2 and this chapter of this SAR describe in detail the shielding structures, systems, and components (SSCs) important to safety.

The shielding-significant SSCs important to safety have been valuated in this chapter and their impact on personnel and public health and safety resulting from operation of an independent spent fuel storage installation (ISFSI) utilizing the HI-STORM FW system has been evaluated.

It has been shown that the design of the shielding system of the HI-STORM FW 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-STORM FW system will allow safe storage of spent fuel in full conformance with 10CFR72.

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## **APPENDIX 5.A**

### **SAMPLE INPUT FILES FOR SAS2H, ORIGEN-S, AND MCNP**

**Withheld in Accordance with 10 CFR 2.390**

## CHAPTER 6<sup>†</sup>: CRITICALITY EVALUATION

### 6.0 INTRODUCTION

This chapter documents the criticality evaluation of the HI-STORM FW system for the storage of spent nuclear fuel in accordance with 10CFR72.124 [6.1.2]. The evaluation shows that the maximum  $k_{\text{eff}}$  value, including all applicable biases and uncertainties is below 0.95 for all normal, off-normal and accident conditions. This demonstrates that the HI-STORM FW system meets the criticality safety requirements of 10CFR72 [6.1.2] and the Standard Review Plan for Dry Cask Storage Systems (NUREG-1536) [6.1.1].

In addition, this chapter describes the HI-STORM FW system design structures and components important to criticality safety and defines the limiting fuel characteristics in sufficient detail to provide a sufficient basis for the evaluation of the package.

Note that the analysis methodologies and modeling assumptions are identical to those utilized in the licensing of the HI-STORM 100 system in Docket No. 72-1014 ([6.0.1], Chapter 6), except for the following:

- A newer version of the Monte Carlo code MCNP, namely MCNP5, is used, together with the corresponding cross-sections. The benchmark calculations were updated accordingly.

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 unquantified 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

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<sup>†</sup> 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. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the Glossary and component nomenclature of the Bill-of-Materials (Section 1.5).

Evaluations and results presented in this chapter are supported by documented calculation package(s) [6.0.2].

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.

## 6.1 DISCUSSION AND RESULTS

In conformance with the principles established in NUREG-1536 [6.1.1] and 10CFR72.124 [6.1.2], the results in this chapter demonstrate that the effective multiplication factor ( $k_{\text{eff}}$ ) of the HI-STORM FW 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, the results demonstrate that the HI-STORM FW 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-STORM FW system when fully loaded with fuel of the highest permissible reactivity.

Criticality safety of the HI-STORM FW system depends on the following four principal design parameters:

1. The inherent geometry of the fuel basket designs within the MPC;
2. The fuel basket structure which is made entirely of the Metamic-HT neutron absorber material;
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 in the PWR fuel basket.

The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 12 have no adverse effect on the design parameters important to criticality safety, except for the non-mechanistic tip-over event, which could result in limited plastic deformation of the basket. However, a bounding basket deformation is already included in the criticality models for normal conditions, and thus, from the criticality safety standpoint, the off-normal and accident conditions are identical to those for normal conditions.

The HI-STORM FW system is designed such that the fixed neutron absorber will remain effective for a storage period greater than 60 years, and there are no credible mechanisms that would cause its loss or a diminution of its effectiveness (see Chapter 8, specifically Section 8.9, and Section 10.1.6.3 for further information on the qualification and testing of the neutron absorber material). 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-STORM FW system does not rely on the use of any of the following aids to the reduction of reactivity present in the storage system:

- burnup of fuel
- fuel-related burnable neutron absorbers
- more than 90 percent of the B-10 content for the Metamic-HT fixed neutron absorber undergirded by comprehensive tests as described in Subsection 10.1.6.3.

The HI-STORM FW system consists of the HI-STORM FW storage cask, the HI-TRAC VW transfer cask and Multi-Purpose-Canisters (MPCs) for PWR and BWR fuel (see Chapter 1, Table 1.0.1). Both the HI-TRAC VW transfer cask and the HI-STORM FW storage cask accommodate the interchangeable MPC designs. The HI-STORM FW storage cask uses concrete as a shield for both gamma and neutron radiation, while the HI-TRAC VW uses lead and steel for gamma radiation and a water-filled jacket for neutron shielding. The design details can be found in the drawing packages in Section 1.5.

While the MPCs are in the HI-STORM FW cask during storage, they are internally dry (no moderator), and thus, the reactivity is very low ( $k_{\text{eff}} \sim 0.6$ ). However, the MPCs are flooded for loading and unloading operations in the HI-TRAC VW cask, which represents the limiting case in terms of reactivity. Therefore, the majority of the analyses have been performed with the MPCs in a HI-TRAC VW cask, and only selected cases have been performed for the HI-STORM FW cask.

Confirmation of the criticality safety of the HI-STORM FW system was accomplished with the three-dimensional Monte Carlo code MCNP5 [6.1.4]. 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 [6.1.5].

To assess the reactivity effects due to temperature changes, CASMO-4, a two-dimensional transport theory code [6.1.6] for fuel assemblies was used. CASMO-4 was not used for quantitative information, but only to qualitatively indicate the direction and approximate magnitude of the reactivity effects.

Benchmark calculations were made to compare the primary code package (MCNP5) with experimental data, using critical experiments selected to encompass, insofar as practical, the design parameters of the HI-STORM FW system. The most important parameters are (1) the enrichment, (2) cell spacing, (3) the  $^{10}\text{B}$  loading of the neutron absorber panels, and (4) the soluble boron concentration in the water (for PWR fuel). The critical experiment benchmarking work is summarized in Appendix 6.A.

To assure the true reactivity will always be less than the calculated reactivity, the following conservative design criteria and assumptions were made:

- The MPCs are assumed to contain the most reactive fresh fuel authorized to be loaded into a specific basket design.
- No credit for fuel burnup is assumed, either in depleting the quantity of fissile nuclides or in producing fission product.
- The fuel stack density is assumed to be at 97.5% of the theoretical density for all criticality analyses. This is a conservative value, since it corresponds to a very high pellet density of 99% or more of the theoretical density. Note that this difference between stack and pellet density is due to the necessary dishing and chamfering of the pellets.
- No credit is taken for the  $^{234}\text{U}$  and  $^{236}\text{U}$  in the fuel.
- When flooded, the moderator is assumed to be water, with or without soluble boron, at a temperature and density corresponding to the highest reactivity within the expected operating range.
- When credit is taken for soluble boron, a  $^{10}\text{B}$  content of 18.0 wt% in boron is assumed.
- Neutron absorption in minor structural members is neglected, i.e., spacer grids are replaced by water. This is conservative since studies presented in Section 6.2.1 show that all assemblies are undermoderated, and that the reduction in the amount of (borated or unborated) water within the fuel assembly always results in a reduction of the reactivity. The presence of any other structural material, which would reduce the amount of water, is therefore bounded by those studies, and neglecting this material is conservative. Additionally, the potential neutron absorption of those materials is neglected.
- 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 with pure unborated water.
- Planar-averaged enrichments are assumed for BWR fuel. Analyses are presented that demonstrate that the use of planar-averaged enrichments is appropriate.
- 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.

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- For evaluation of the bias and bias uncertainty, two approaches are utilized. One where the results of the benchmark calculations are used directly and one where benchmark calculations that result in a  $k_{\text{eff}}$  greater than 1.0 are conservatively truncated to 1.0000. Consistent with NUREG-1536, the larger of the combined bias and bias uncertainty of the two approaches is used.
- The water reflector above and below the fuel is assumed to be unborated water, even if borated water is used in the fuel region.
- 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 undamaged fuel assemblies, as defined in the Glossary, all fuel rod positions are assumed to contain a fuel rod. To qualify assemblies with missing fuel rods, those 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.
- For DFCs, a large ID and small wall thickness is used. This is conservative, since it maximizes the area of the optimum moderated fuel, and minimizes the neutron absorption in the DFC wall.

The design basis criticality safety calculations are performed for a single internally flooded HI-TRAC VW transfer cask with full water reflection on all sides (limiting cases for the HI-STORM FW system), for fuel assemblies listed in Chapter 2, are conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.3), and include the calculational bias, uncertainties, and calculational statistics. In addition, a few results for single internally dry (no moderator) HI-STORM FW storage casks with full water reflection on all external surfaces of the overpack, including the annulus region between the MPC and overpack, are performed to confirm the low reactivity of the HI-STORM FW system in storage.

Note that throughout this chapter reactivity results are stated as maximum neutron multiplication factor values ( $k_{\text{eff}}$ ) conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.3), and including the calculational bias, uncertainties, and calculational statistics, unless otherwise noted.

For undamaged fuel, and for each of the MPC designs under flooded conditions (HI-TRAC VW), minimum soluble boron concentration (if applicable) and fuel assembly classes<sup>††</sup>, Tables

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<sup>††</sup> The assembly classes for BWR and PWR fuel are defined in Section 6.2.

6.1.1, 6.1.2 and 6.1.6 list the bounding maximum  $k_{\text{eff}}$  value, and the associated maximum allowable enrichment. Tables 6.1.1 and 6.1.2 provide the information for undamaged fuel without known or suspected cladding defects larger than pinhole leaks or hairline cracks, while Table 6.1.6 provides the information for low-enriched, channeled BWR undamaged fuel without known or suspected grossly breached fuel rods. The maximum allowed enrichments and the minimum soluble boron concentrations are also cited in Subsection 2.1.

For MPCs in the HI-STORM FW under dry conditions, results are listed in Table 6.1.3 for selected assembly classes.

For MPCs loaded with a combination of undamaged and damaged fuel assemblies under flooded conditions, results are listed in Tables 6.1.4 and 6.1.5. For each of the MPC designs, the tables indicate the maximum number of DFCs and list the fuel assembly classes, the bounding maximum  $k_{\text{eff}}$  value, the associated maximum allowable enrichment, and if applicable the minimum soluble boron concentration. Allowed enrichments are also cited in Subsection 2.1.

These results confirm that the maximum  $k_{\text{eff}}$  values for the HI-STORM FW system are below the limiting design criteria ( $k_{\text{eff}} < 0.95$ ) 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-STORM FW system to safely accommodate damaged fuel and fuel debris is demonstrated in Subsection 6.4.4. The capability of the HI-STORM FW to accommodate low enriched, channeled BWR fuel as undamaged fuel demonstrated in Subsection 6.4.9.

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-STORM FW System for storage is dry (no moderator) and the reactivity is very low. For arrays of HI-STORM FW storage 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 PWR fuel in the MPC-37, soluble boron in the water is credited. There is a strict administrative control on the soluble boron concentration during loading and unloading of the MPC, consisting of frequent and independent measurements (For details see Subsections 9.2.2, 9.2.3, 9.2.4, and 9.4.3 and the bases for LCO 3.3.1 in Chapter 13). An accidental loss of soluble boron is therefore not credible and hence not considered.

TABLE 6.1.1

BOUNDING MAXIMUM  $k_{\text{eff}}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-37  
(HI-TRAC VW)

Fuel Assembly Class	4.0 wt% $^{235}\text{U}$ Maximum Enrichment <sup>†</sup>		5.0 wt% $^{235}\text{U}$ Maximum Enrichment <sup>†</sup>	
	Minimum Soluble Boron Concentration (ppm)	Maximum $k_{\text{eff}}$	Minimum Soluble Boron Concentration (ppm)	Maximum $k_{\text{eff}}$
14x14A	1000	0.8946	1500	0.8983
14x14B	1000	0.9213	1500	0.9282
14x14C	1000	0.9211	1500	0.9277
15x15B	1500	0.9129	2000	0.9311
15x15C	1500	0.9029	2000	0.9188
15x15D	1500	0.9223	2000	0.9421
15x15E	1500	0.9206	2000	0.9410
15x15F	1500	0.9244	2000	0.9455
15x15H	1500	0.9142	2000	0.9325
15x15I	1500	0.9155	2000	0.9362
16x16A	1000	0.9275	1500	0.9366
17x17A	1500	0.9009	2000	0.9194
17x17B	1500	0.9181	2000	0.9380
17x17C	1500	0.9222	2000	0.9424
17x17D	1500	0.9183	2000	0.9384
17x17E	1500	0.9203	2000	0.9392

<sup>†</sup> For maximum allowable enrichments between 4.0 wt%  $^{235}\text{U}$  and 5.0 wt%  $^{235}\text{U}$ , the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified for each assembly class.

TABLE 6.1.2

BOUNDING MAXIMUM  $k_{eff}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-89  
(HI-TRAC VW)

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% $^{235}\text{U}$ )	Maximum $k_{eff}$
7x7B	4.8	0.9317
8x8B	4.8	0.9369
8x8C	4.8	0.9399
8x8D	4.8	0.9380
8x8E	4.8	0.9281
8x8F	4.5	0.9328
9x9A	4.8	0.9421
9x9B	4.8	0.9410
9x9C	4.8	0.9338
9x9D	4.8	0.9342
9x9E/F	4.5	0.9346
9x9G	4.8	0.9307
10x10A	4.8	0.9435
10x10B	4.8	0.9417
10x10C	4.8	0.9389
10x10F	4.7	0.9440
10x10G	4.6	0.9466

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TABLE 6.1.3

REPRESENTATIVE  $k_{\text{eff}}$  VALUES FOR MPC-37 AND MPC-89 IN THE HI-STORM FW OVERPACK

<b>MPC</b>	<b>Fuel Assembly Class</b>	<b>Maximum Allowable Planar-Average Enrichment (wt% <sup>235</sup>U)</b>	<b>Maximum <math>k_{\text{eff}}</math></b>
MPC-37	17x17B	5.0	0.6076
MPC-89	10x10A	4.8	0.3986

TABLE 6.1.4  
 BOUNDING MAXIMUM  $k_{\text{eff}}$  VALUES FOR THE MPC-37  
 WITH UP TO 12 DFCs

Fuel Assembly Class of Undamaged Fuel	4.0 wt% $^{235}\text{U}$ Maximum Enrichment for Undamaged Fuel and Damaged Fuel/Fuel Debris <sup>†</sup>		5.0 wt% $^{235}\text{U}$ Maximum Enrichment for Undamaged Fuel and Damaged Fuel/Fuel Debris <sup>†</sup>	
	Minimum Soluble Boron Concentration (ppm)	Maximum $k_{\text{eff}}$	Minimum Soluble Boron Concentration (ppm)	Maximum $k_{\text{eff}}$
All 14x14, 16x16A	1300	0.9023	1800	0.9163
All 15x15, all 17x17	1800	0.9032	2300	0.9276

<sup>†</sup> For maximum allowable enrichments between 4.0 wt%  $^{235}\text{U}$  and 5.0 wt%  $^{235}\text{U}$ , the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified for each assembly class.

TABLE 6.1.5

BOUNDING MAXIMUM  $k_{\text{eff}}$  VALUES FOR THE MPC-89  
WITH UP TO 16 DFCs

<b>Fuel Assembly Class</b>	<b>Maximum Allowable Planar-Average Enrichment (wt% <math>^{235}\text{U}</math>)</b>	<b>Maximum <math>k_{\text{eff}}</math></b>
All BWR Classes except 8x8F, 9x9E/F, 10x10F and 10x10G	4.8	0.9464
8x8F, 9x9E/F and 10x10G	4.0	0.9299
10x10F	4.6	0.9428

TABLE 6.1.6

BOUNDING MAXIMUM  $k_{\text{eff}}$  VALUES FOR THE MPC-89  
WITH LOW ENRICHED, CHANNELED BWR FUEL

<b>Fuel Assembly Class</b>	<b>Maximum Allowable Planar-Average Enrichment (wt% <math>^{235}\text{U}</math>)</b>	<b>Maximum <math>k_{\text{eff}}</math></b>
All BWR Classes	3.3	0.9325

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## 6.2 SPENT FUEL LOADING

Due to the large number of minor variations in the fuel assembly dimensions, the use of explicit dimensions in defining the authorized contents could limit or complicate the applicability of the HI-STORM FW system. To resolve this limitation, a number of fuel assembly classes for both fuel types (PWR and BWR) are defined based on bounding fuel dimensions. The results of parametric studies justify using those bounding fuel dimensions for defining the authorized contents.

### 6.2.1 Definition of Assembly Classes

For each array size 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; and (3) number and locations of guide tubes (PWR) or water rods (BWR). The assembly classes for PWR and BWR fuel are defined in Chapter 2, Tables 2.1.2 and 2.1.3, respectively. It should be noted that these assembly classes are consistent with the class designations in the HI-STORM 100 FSAR (Docket No. 72-1014). Specifically, assembly classes with the same identifier refer to the same set of limiting dimensions. However, some classes have been removed and others have been added compared to the HI-STORM 100.

In HI-STORM 100 FSAR (Docket No. 72-1014), extensive analyses of fuel dimensional variations have been performed. These calculations demonstrate that the maximum reactivity corresponds to:

- maximum active fuel length,
- maximum fuel pellet diameter,
- maximum fuel rod pitch,
- minimum cladding outside diameter (OD),
- maximum cladding inside diameter (ID),
- minimum guide tube/water rod thickness, and
- maximum channel thickness (for BWR assemblies only).

The reason that those are bounding dimensions, i.e. that they result in maximum reactivity is directly based on, and can be directly derived from the three main characteristics affecting reactivity, namely 1) characteristics of the fission process; 2) the characteristics of the fuel assemblies and 3) the characteristics of the neutron absorber in the basket. These affect the reactivity as follows:

- The neutrons generated by fission are fast neutrons while the neutrons that initiate the fission need to be thermal neutrons. A moderator (water) is therefore necessary for the nuclear chain reaction to continue.



- Fuel assemblies are predominantly characterized by the amount of fuel and the fuel-to-water (moderator) ratio. Increasing the amount of fuel, or the enrichment of the fuel, will increase the amount of fissile material, and therefore increase reactivity. Regarding the fuel-to-water ratio, it is important to note that commercial PWR and BWR assemblies are undermoderated, i.e. they do not contain enough water for a maximum possible reactivity.
- The neutron poison in the basket walls uses B-10, which is an absorber of thermal neutrons. This poison therefore also needs water (moderator) to be effective. This places a specific importance on the amount of water between the outer rows of the fuel assemblies and the basket cell walls. Note that this explains some of the differences in reactivity between the different assembly types in the same basket, even for the same enrichment, where assemblies with a smaller cross section, i.e. which have more water between the periphery of the assembly and the surrounding wall, generally have a lower reactivity.

Based on these characteristics, the following conclusions can be made:

- Since fuel assemblies are undermoderated, any changes in geometry inside the fuel assembly that increases the amount of water while maintaining the amount of fuel are expected to increase reactivity. This explains why reducing the cladding or guide tube/water rod thicknesses, or increasing the fuel rod pitch results in an increase in reactivity.
- Increasing the active length will increase the amount of fuel while maintaining the fuel-to-water ratio, and therefore increase reactivity.
- The channel of the BWR assembly is a structure located outside of the rod array. It therefore does not affect the water-to-fuel ratio within the assembly. However, it reduces the amount of water between the assembly and the neutron poison, therefore reducing the effective thermalization for the poison. Therefore, an increase of the channel wall thickness will increase reactivity.
- In respect to the effect of the fuel pellet diameter, several compensatory effects need to be considered. Increasing the diameter will tend to increase the reactivity due to the increase in the fuel amount. However, it will also change the fuel-to-water-ratio, and will therefore make the fuel more undermoderated, which in turn tends to reduce reactivity. The effect of this change in moderation may depend on the condition of the pellet-to-clad gap. Assuming an empty pellet-to clad gap, which would be consistent with undamaged fuel rods, the change in moderation is small, and the net effect is an increase in reactivity, since the effect of the increase in the fissionable material dominates. In this case, the maximum pellet diameter is more reactive. When the pellet-to-clad gap is conservatively flooded, as recommended by NUREG 1536 (see section 6.4.2.3), a reduction of the fuel pellet diameter will also result in an increase in the amount of water, i.e. have a double effect on the water-to-fuel ratio. In this case, it is possible that a slight reduction may result in no reduction or even an increase in reactivity. However, this is caused by a further amplification of the conservative assumption of the flooded pellet-to-clad gap, not

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by a positive increase in reactivity from the reduction in fuel (which would be counter-intuitive). Therefore, in order not to overstate the conservative effect of the flooded fuel-to-clad gap, the calculations for the variation of the fuel pellet diameter are performed for a flooded gap of constant thickness by also changing the clad ID.

Since all assemblies have the same principal design, i.e. consist of bundles of clad fuel rods, most of them with embedded guide/instrument tubes or water rods or channels, the above conclusions apply to all of them, and the bounding dimensions are therefore also common to all fuel assemblies analyzed here. Nevertheless, to clearly demonstrate that the main assumption is true, i.e. that all assemblies are undermoderated, a study was performed for all assembly types where the pellet-to-clad gap is empty instead of being flooded (a conservative assumption for the design basis calculations, see Section 6.4.2.3) The results are listed in Table 6.2.3, in comparison with the results of the reference cases with the flooded gap from Section 6.1 for those assembly types. In all cases, the reactivity is reduced compared to the reference case. This verifies that all assembly types considered here are in fact undermoderated, and therefore validates the main assumption stated above. All assembly types are therefore behaving in a similar fashion, and the bounding dimensions are therefore applicable to all assembly types. This discussion and the corresponding conclusions not only affect fuel behavior, but also other moderation effects, and is therefore further referenced in Section 6.3.1 and 6.4.2

As a result, the authorized contents in Subsection 2.1 are defined in terms of those bounding assembly parameters for each class.

Nevertheless, to further demonstrate that the aforementioned characteristics are in fact bounding for the HI-STORM FW, parametric studies were performed on reference PWR and BWR assemblies, namely PWR assembly class 17x17B and BWR assembly class 10x10A. The results of these studies are shown in Table 6.2.1 and 6.2.2, and verify the bounding parameters listed above. Note that in the studies presented in Tables 6.2.1 and 6.2.2, the fuel pellet diameter and cladding inner diameter are changed together. This is to keep the cladding-to-pellet gap, which is conservatively flooded with pure water in all cases (see Section 6.4.2.3), at a constant thickness, to ensure the studies evaluate the fuel parameters rather than the moderation conditions, as discussed above.

In addition to those dimensions, additional fuel assembly characteristics important to criticality control are the location of guide tubes, water rods, part length rods, and rods with differing dimensions (classes 9x9E/F only). These are identified in the assembly cross sections provided in Appendix 6.B, Section B.4.

In all cases, the gadolinia ( $Gd_2O_3$ ) normally incorporated in BWR fuel, and Integral Fuel Burnable Absorbers (IFBA) used in PWR fuel was conservatively neglected.

Some assembly classes contain partial length rods. There are differences in location of those partial length rods within the assembly that influence how those rods affect reactivity: Assembly classes 9x9A, 10x10A, 10x10B and 10x10F have partial length rods that are completely

surrounded by full length rods, whereas assembly class 10x10G has those partial length rods on the periphery of the assembly or facing the water gap, where they directly only face two full length rods (see Appendix 6.B, Section B.4). To determine a bounding configuration for those assembly classes where partial length rods are completely surrounded by full length rods, calculations are listed in Table 6.2.2 for the actual (real) assembly configuration and for the axial segments (assumed to be full length) with and without the partial length rods. The results show that the configurations with only the full length rods present, i.e. where the partial length rods are assumed completely absent from the assembly, is bounding. This is an expected outcome, since LWR assemblies are typically undermoderated, therefore reducing the fuel-to-water-ratio within the rod array tends to increase reactivity. Consequently, all assembly classes that contain partial length rods surrounded by full-length rods are analyzed with the partial length rods absent. For assembly class 10x10G, calculations with different assumptions for the length of the part-length rods are presented in Table 6.2.7, and show that reducing the length of the part length rods reduces reactivity. This means that the reduction in the fuel amount is more dominating than the change in moderation for this configuration. For this class, all rods therefore are assumed full length. Note that in neither of the cases is the configuration with the actual part length rods bounding. The specification of the authorized contents has therefore no minimum requirement for the active fuel length of the partial length rods.

BWR assemblies are specified in Table 2.1.3 with a maximum planar-average enrichment. The analyses presented in this chapter use a uniform enrichment, equal to the maximum planar-average. Analyses presented in the HI-STORM FSAR ([6.0.1], Chapter 6, Appendix 6.B) show that this is a conservative approach, i.e. that a uniform enrichment bounds the planar-average enrichment in terms of the maximum  $k_{\text{eff}}$ . To verify that this is applicable to the HI-STORM FW, those calculations were re-performed in the MPC-89. The results are presented in Table 6.2.4, and show that, as expected, the planar average enrichments bound or are statistically equivalent to the distributed enrichment in the HI-STORM FW as they do in the HI-STORM 100. To confirm that this is also true for the higher enrichments analyzed here, additional calculations were performed and are presented in Table 6.2.2 in comparison with the results for the uniform enrichment. Since the maximum planar-average enrichment of 4.8 wt%  $^{235}\text{U}$  is above the actual enrichments of those assemblies, actual (as-built) enrichment distributions are not available. Therefore, several bounding cases are analyzed. Note that since the maximum planar-average enrichment of 4.8 wt%  $^{235}\text{U}$  is close to the maximum rod enrichment of 5.0 wt%  $^{235}\text{U}$ , the potential enrichment variations within the cross section are somewhat limited. To maximize the differences in enrichment under these conditions, the analyzed cases assume that about 50% of the rods in the cross section are at an enrichment of 5.0 wt%  $^{235}\text{U}$ , while the balance of the rods are at an enrichment of about 4.6 wt%, resulting in an average of 4.8 wt%. Calculations are performed for cross sections where all full-length and part-length, or only all full-length rods are present. For each case, two conditions are analyzed that places the different enrichment in areas with different local fuel-to-water ratios. Specifically, one condition places the higher enriched rods in locations where they are more surrounded by other rods, whereas the other condition places them in locations where they are more surrounded by water, such as near the water-rods or the periphery of the assembly. The results are also included in table 6.2.2 and show that in all cases, the maximum  $k_{\text{eff}}$  calculated for the distributed enrichments are statistically equivalent to

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or below those for the uniform enrichments. Therefore, modeling BWR assemblies with distributed enrichments using a uniform enrichment equal to the planar-average value is acceptable and conservative. The assumed enrichment distributions analyzed are shown in Appendix 6.B.

Note that for some BWR fuel assembly classes, the Zircaloy water rod tubes are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the specification of the authorized contents. For these cases, the bounding water rod thickness is listed as zero.

Two BWR classes (8x8B and 8x8D) are specified with slight variation in the number of fuel and/or water rods (see Section 6.B.4). The results listed in Section 6.1 utilize the minimum number of fuel rods, i.e. maximizing the water-to-fuel ratio. To show that this is appropriate and bounding, calculations were also performed with the alternative configurations, and are presented in Table 6.2.5. The results show that the reference conditions used for the calculations documented in Section 6.1 are in fact bounding.

For BWR assembly class 9x9E/F, two patterns of water rods were analyzed (see Section 6.B.4). The comparison is also presented in Table 6.2.5 and shows that the condition with the larger water rod spacing is bounding.

For PWR assembly class 15x15I (see Section 6.B.4), calculations with and without guide rods were performed. The comparison is also presented in Table 6.2.5. The case without the guide rods is used as the design basis case for this assembly type, therefore, no specific restrictions on the location and number of guide rods exists.

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 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 of annular pellets at the top and bottom performed for the HI-STORM FW with various pellet IDs (see Table 6.2.6) 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. This is consistent with the HI-STORM 100, where the same analyzed conditions are analyzed and qualified.

TABLE 6.2.1

REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS in PWR Fuel in the  
MPC-37 with 2000 ppm soluble boron concentration  
(all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	Maximum $k_{eff}$	standard deviation
17x17B (5.0 wt% Enrichment)	Reference	0.9374	0.0004
increase pellet OD and clad ID (+0.004)	0.0052	0.9426	0.0003
decrease pellet OD and Clad ID (-0.004)	-0.0058	0.9316	0.0004
increase clad OD (+0.004)	-0.0014	0.9360	0.0004
decrease clad OD (-0.004)	0.0017	0.9391	0.0004
increase guide tube thickness (+0.004)	-0.0001	0.9373	0.0004
decrease guide tube thickness (-0.004)	0.0004	0.9378	0.0003
remove guide tubes (i.e., replace the guide tubes with water)	0.0009	0.9383	0.0004
reduced active length (100 Inches)	-0.0020	0.9354	0.0004
increase rod pitch (+0.004)	0.0019	0.9393	0.0004
reduce rod pitch (-0.004)	-0.0017	0.9357	0.0004

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TABLE 6.2.2

REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS for BWR Fuel in the  
MPC-89  
(all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	Maximum $k_{eff}$	standard deviation
10x10A (Reference, full-length rods only)	Reference	0.9429	0.0004
increase pellet OD and Clad ID (+0.004)	0.0037	0.9466	0.0004
decrease pellet OD and Clad ID (-0.004)	-0.0042	0.9387	0.0004
increase clad OD (+0.004)	-0.0021	0.9408	0.0003
decrease clad OD (-0.004)	0.0032	0.9461	0.0004
increase water rod thickness (+0.004)	0.0002	0.9431	0.0004
decrease water rod thickness (-0.004)	0.0009	0.9438	0.0004
remove water rods (i.e., replace the water rod tubes with water)	0.0031	0.9460	0.0004
reduced active length (100 Inches)	-0.0026	0.9403	0.0004
remove channel	-0.0113	0.9316	0.0003
increase channel thickness (+0.020)	0.0007	0.9436	0.0003
full-length and part-length rods (real assembly)	-0.0054	0.9375	0.0004
part-length rods extended to full-length	-0.0102	0.9327	0.0004
increased rod pitch (+0.004)	0.0050	0.9479	0.0004
reduced rod pitch (-0.004)	-0.0043	0.9386	0.0003
distributed enrichment, Case 1	-0.0011	0.9418	0.0003
distributed enrichment, Case 2	+0.0004	0.9433	0.0003
distributed enrichment, Case 3	-0.0099	0.9330	0.0004
distributed enrichment, Case 4	-0.0121	0.9308	0.0003

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TABLE 6.2.3

## EFFECT OF THE FLOODING OF THE PELLETT-TO-CLAD GAP

Fuel Assembly Class	Maximum $k_{\text{eff}}$ at 5.0 wt% $^{235}\text{U}$ Maximum Enrichment		
	Flooded Pellet-to-Clad Gap	Empty Pellet-to-Clad Gap	Difference
14x14A	0.8983	0.8962	-0.0021
14x14B	0.9282	0.9235	-0.0047
14x14C	0.9277	0.9237	-0.0038
15x15B	0.9311	0.9284	-0.0027
15x15C	0.9188	0.9164	-0.0024
15x15D	0.9421	0.9386	-0.0035
15x15E	0.9410	0.9371	-0.0039
15x15F	0.9455	0.9408	-0.0047
15x15H	0.9325	0.9300	-0.0025
15x15I	0.9357	0.9305	-0.0052
16x16A	0.9366	0.9284	-0.0082
17x17A	0.9194	0.9160	-0.0034
17x17B	0.9380	0.9335	-0.0045
17x17C	0.9424	0.9375	-0.0049
17x17D	0.9384	0.9323	-0.0061
17x17E	0.9392	0.9346	-0.0046

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TABLE 6.2.3 (continued)

## EFFECT OF THE FLOODING OF THE PELLETT-TO-CLAD GAP

Fuel Assembly Class	Maximum $k_{eff}$		
	Flooded Pellet-to-Clad Gap	Empty Pellet-to-Clad Gap	Difference
7x7B	0.9317	0.9261	-0.0056
8x8B	0.9369	0.9318	-0.0051
8x8C	0.9399	0.9331	-0.0068
8x8D	0.9380	0.9334	-0.0046
8x8E	0.9281	0.9230	-0.0051
8x8F	0.9328	0.9275	-0.0053
9x9A	0.9421	0.9370	-0.0051
9x9B	0.9410	0.9292	-0.0118
9x9C	0.9338	0.9290	-0.0048
9x9D	0.9342	0.9294	-0.0048
9x9E/F	0.9346	0.9261	-0.0085
9x9G	0.9307	0.9250	-0.0057
10x10A	0.9435	0.9391	-0.0044
10x10B	0.9417	0.9317	-0.0100
10x10C	0.9389	0.9333	-0.0056
10x10F	0.9440	0.9395	-0.0045
10x10G	0.9466	0.9408	-0.0058

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Table 6.2.4

COMPARISON CALCULATIONS FOR BWR FUEL WITH AVERAGE AND  
DISTRIBUTED ENRICHMENTS

Case	Planar Average Enrichment (wt%)	Peak Rod Enrichment (wt%)	Maximum $k_{eff}$	
			Planar Average Enrichment (wt%)	Peak Rod Enrichment (wt%)
8x8C	3.01	3.80	0.8358	0.8309
8x8C	3.934	4.9	0.8975	0.8899
8x8D	3.42	3.95	0.8628	0.8636
8x8D	3.78	4.40	0.8862	0.8855
8x8D	3.90	4.90	0.8934	0.8913
9x9B	4.34	4.71	0.9195	0.9179
9x9D	3.35	4.34	0.8575	0.8456
Hypothetical #1 (48 outer rods of 3.967%E, 14 inner rods of 5.0%)	4.20	5.00	0.9104	0.9102
Hypothetical #2 (48 outer rods of 4.354%E, 14 inner rods of 5.0%)	4.50	5.00	0.9258	0.9247

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Table 6.2.5

VARIATIONS OF NUMBER OF FUEL AND/OR WATER RODS FOR ASSEMBLY  
CLASSES 8x8B AND 8x8D (see Appendix B)

Case	Maximum $k_{eff}$
Assembly Class 8x8B	
63 Fuel Rods (Reference)	0.9369
64 Fuel Rods	0.9342
Assembly Class 8x8D	
60 Fuel Rods, no water rods modeled (Reference)	0.9380
60 Fuel Rods, 2 larger, 2 smaller water rods	0.9362
60 Fuel Rods, 4 larger water rods	0.9347
60 Fuel Rods, 4 smaller water rods	0.9359
60 Fuel Rods, 1 large water rods	0.9343
61 Fuel Rods, 3 water rods	0.9354

VARIATION OF WATER ROD LOCATIONS FOR ASSEMBLY CLASS 9x9E/F (see  
Appendix B)

Case	Maximum $k_{eff}$
Adjacent Water Rods (Reference)	0.9346
Water Rods separated by a Fuel Rod	0.9313

VARIATION OF GUIDE RODS FOR ASSEMBLY CLASS 15x15I (see Appendix B)

Case	Maximum $k_{eff}$
No Guide Rods (Reference)	0.9362
8 Guide Rods	0.9259

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Table 6.2.6

## EFFECT OF ANNULAR PELLETS IN THE TOP AND BOTTOM 12 INCHES OF THE ACTIVE REGION

<b>Diameter of Annulus in Pellets</b>	<b>Maximum <math>k_{eff}</math></b>
Assembly Class 17x17B, 5% Enrichment, Undamaged Fuel	
None (Reference)	0.9380
0.1 Inches	0.9382
0.2 Inches	0.9379
0.3 Inches	0.9371
0.4 Inches	0.9371
0.5 Inches	0.9363
0.6 Inches	0.9368
Assembly Class 16x16A, 5% Enrichment, Damaged and Undamaged Fuel	
None (Reference)	0.9163
0.2 Inches	0.9162
Assembly Class 15x15F, 5% Enrichment, Damaged and Undamaged Fuel	
None (Reference)	0.9276
0.2 Inches	0.9266

Table 6.2.7

## EFFECT OF PARTIAL LENGTH RODS FOR ASSEMBLY CLASS 10x10G

<b>Length of Partial Length Rods as a Percentage of Full Length Rods</b>	<b>Maximum <math>k_{eff}</math></b>
100%	0.9466
75%	0.9455
50%	0.9404
25%	0.9286
0%	0.9208

## 6.3 MODEL SPECIFICATION

### 6.3.1 Description of Calculational Model

Figures 6.3.1 through 6.3.5 show representative cross sections of the criticality models for the two baskets. Figures 6.3.1 and 6.3.2 show a single cell from each of the two baskets. Figures 6.3.3 and 6.3.4 show the entire MPC-37 and MPC-89 basket, respectively. Figure 6.3.5 shows a sketch of the calculational model in the axial direction.

Full three-dimensional calculational models were used for all calculations. The calculational models explicitly define the fuel rods and cladding, the guide tubes, water rods and the channel (for the BWR assembly), the neutron absorber walls of the basket cells, and the surrounding MPC shell and overpack. For the flooded condition (loading and unloading), pure, unborated water was assumed to be present in the fuel rod pellet-to-clad gaps, since this represents the bounding condition as demonstrated in Section 6.4.2.3. Appendix 6.B provides sample input files for typical MPC basket designs

Note that the water thickness above and below the fuel is modeled as unborated water, even when borated water is present in the fuel region.

The discussion provided in Section 6.2.1 regarding the principal characteristics of fuel poison is also important for the various studies presented in this section, and supports the fact that those studies only need to be performed for a single BWR and PWR assembly type, and that the results of those studies are then generally applicable to all assembly types. The studies and the relationship to the discussion in Section 6.2.1 are listed below. Note that this approach is consistent with that used for the HI-STORM 100.

**Basket Manufacturing Tolerance:** The two aspects of the basket tolerance that are evaluated are the cell wall thickness and the cell ID. The reduced cell wall thickness results in a reduced amount of poison (since the material composition of the wall is fixed), and therefore in an increase in reactivity. The reduced cell ID reduces amount of water between the fuel the poison, and therefore the effectiveness of the poison material. Both effects are simply a function of the geometry, and are independent of the fuel type.

**Panel Gaps:** Similar to the basket manufacturing tolerance for the cell wall thickness, this tolerance has a small effect on the overall poison amount of the basket, which would affect the reactivity of the system independent of the fuel type.

**Eccentric positioning (see Section 6.3.3):** When a fuel assembly is located in the center of a basket cell, it is surrounded by equal amounts of water on all sides, and hence the thermalization of the neutrons that occur between the assembly and the poison in the cell wall, and hence the effectiveness of the poison, is also equal on all sides. For an eccentric positioning, the

effectiveness of the poison is now reduced on those sides where the assembly is located close to the cell walls, and increased on the opposite sides. This creates a compensatory situation for a single cell, where the net effect is not immediately clear. However, for the entire basket, and for the condition where all assemblies are located closest to the center of the basket, the four assemblies at the center of the basket are now located close to each other, separated by poison plates with a reduced effectiveness since they are not surrounded by water on any side. This now becomes the dominating condition in terms of reactivity increase. This effect is also applicable to all assembly types, since those assemblies are all located close to the center of the basket, i.e. the eccentric position with all assemblies moved towards the center will be bounding regardless of the assembly type.

The basket geometry can vary due to manufacturing tolerances and due to potential deflections of basket walls as the result of accident conditions. The basket tolerances are defined on the drawings in Chapter 1. The structural acceptance criteria for the basket during accident conditions is that the permanent deflection of the basket panels is limited to a fraction of 0.005 (0.5%) of the panel width (see Chapter 3). The analyses in Chapter 3 demonstrate that permanent deformations of the basket walls during accident conditions are far below this limit. In fact, the analyses show that the vast majority of the basket panels remain elastic during and after an accident, and therefore show no permanent deflection whatsoever, and that any deformation is limited to small localized areas. Nevertheless, it is conservatively assumed that 2 adjacent cell walls in each cell are deflected to the maximum extent possible over their entire length and width, i.e. that the cell ID is reduced by 0.5% of the cell width, or 0.045" for the MPC-37 cells and 0.030" for the MPC-89 cells. Stated differently, the minimum cell ID based on tolerances was further reduced by the amounts stated above for all cells in each basket to account for the potential deflections of basket walls during accident conditions. Assuming that all cell sizes are reduced is a simplifying, but very conservative assumption, since cell walls are shared between neighboring cells, so while the deflection of a basket wall would reduce the cell size on one side, it necessarily increases that on the other side of the wall. MCNP5 was used to determine the manufacturing tolerances and deflections that produced the most adverse effect on criticality. After the reactivity 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 conditions in the direction which would increase reactivity. For simplification, the same worst case conditions are used for both normal and accident conditions. For all calculations, fuel assemblies were assumed to be eccentrically located in the cells, since this results in higher reactivities (see Section 6.3.3). Maximum  $k_{\text{eff}}$  results (including the bias, uncertainties, or calculational statistics), along with the selected dimensions, for a number of dimensional combinations are shown in Table 6.3.2 for both baskets. The cell ID is evaluated for minimum (tolerance only), minimum with deformation, nominal and an increased value. The wall thickness is evaluated for nominal and minimum values.

Based on the calculations, the conservative dimensional assumptions listed in Table 6.3.3 were determined for the basket designs. Because the reactivity effect (positive or negative) of the manufacturing tolerances is not assembly dependent, these dimensional assumptions were employed for all criticality analyses.

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The basket is manufactured from individual slotted panels. The panels are expected to be in direct contact with each other (see Drawings in Chapter 1). However, to show that small gaps between panels would have essentially no effect on criticality, calculations are performed with a postulated 0.06" gap between panels, repeated in the axial direction every 10" in all panels. Since it is expected that the effect of these gaps would be small, these calculations were performed with a larger number of particles per cycle, larger number of inactive cycles, and a larger total number of cycles to improve the statistics of each run, so the real reactivity effect could be better separated from the statistical "noise". The results are summarized in Tables 6.3.6 and show that the METAMIC gap has a very small effect. Therefore, all calculations are performed without any gaps between panels.

Variations of water temperature in the cask were analyzed using CASMO-4. The analyses were performed for the assembly class 10x10A in the MPC-89, and for the assembly class 17x17B with 2000 ppm soluble boron in the water in the MPC-37. These are the same assemblies and conditions used for the fuel dimension studies in Section 6.2, and shown there to be representative of all assemblies qualified for those baskets. The results are presented in Table 6.3.1, and show that the minimum water temperature (corresponding to a maximum water density) are bounding. This condition is therefore used in all further calculations. This is expected since an increased temperature results in a reduced water density, a condition that is shown in Section 6.4 to result in reduced reactivities.

Calculations documented in Chapter 3 show that the baskets stay within the applicable structural limits during all normal and accident conditions. Furthermore, the neutron poison material is an integral and non-removable part of the basket material, and its presence is therefore not affected by the accident conditions. Except for the potential deflection of the basket walls that is already considered in the criticality models, damage to the cask under accident conditions is limited to possible loss of the water in the water jacket of the HI-TRAC VW. However, this condition is already considered in the calculational models. Other parameters important to criticality safety are fuel type and enrichment, which are not affected by the hypothetical accident conditions. The calculational models of the cask and basket for the accident conditions are therefore identical to the models for normal conditions, and no separate models need to be developed for accident conditions.

### 6.3.2 Cask Regional Densities

Composition of the various components of the principal designs of the HI-STORM FW system are listed in Table 6.3.4. The cross section set for each nuclide is listed in Table 6.3.8, and is consistent with the cross section sets used in the benchmarking calculations documented in Appendix A. Note that these are the default cross sections chosen by the code.

The HI-STORM FW system is designed such that the fixed neutron absorber will remain effective for a storage period greater than 60 years, and there are no credible means to lose it.

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The continued efficacy of the fixed neutron absorber is assured by acceptance testing, documented in Subsection 10.1.6.3, to validate the  $^{10}\text{B}$  (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 evaluation of the number of neutrons absorbed in the  $^{10}\text{B}$  was performed. The calculation conservatively assumed a constant neutron source for 60 years equal to the initial source for the design basis fuel, as determined in Section 5.2, and shows that the fraction of  $^{10}\text{B}$  atoms destroyed is less than  $10^{-7}$  in 60 years. Thus, the reduction in  $^{10}\text{B}$  concentration in the fixed neutron absorber by neutron absorption is negligible. 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.

### 6.3.3 Eccentric Positioning of Assemblies in Fuel Storage Cells

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. The calculations in this subsection serve to identify the eccentric positioning of assemblies in the fuel storage locations, which results in a higher maximum  $k_{\text{eff}}$  value than the centered positioning. For the cases where the eccentric positioning results in a higher maximum  $k_{\text{eff}}$  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.

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;
- 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 phenomenon that could move all assemblies within a basket consistently to the center or periphery. However, since the configurations listed above bound all credible configurations, they are conservatively used in the analyses.

In Table 6.3.5, results are presented for all representative conditions. The table shows the maximum  $k_{\text{eff}}$  value for centered and the two eccentric configurations for each condition, and the difference in  $k_{\text{eff}}$  between the centered and eccentric positioning. In all cases, moving the assemblies and DFCs to the periphery of the basket results in a reduction in reactivity, compared to the cell centered position, and moving the assemblies and DFCs towards the center results in



an increase in reactivity, compared to the cell centered position. All calculations are therefore performed with assemblies/DFCs moved towards the center of the basket.

TABLE 6.3.1

## CASMO-4 CALCULATIONS FOR EFFECT OF TEMPERATURE

Change in Nominal Parameter	$\Delta k$ Maximum Tolerance		Action/Modeling Assumption
	MPC-37, 17x17B, 5.0 wt%, Borated Water with 2000 ppm Soluble Boron	MPC-89, 10x10A, 4.8 wt%, Fresh Water	
Increase in Temperature			Assume 20°C
20°C	Ref.	Ref.	
40°C	-0.0008	-0.0035	
70°C	-0.0023	-0.0100	
100°C	-0.0042	-0.0180	
10% Void in Moderator			Assume no void
20°C with no void	Ref.	Ref.	
20°C	-0.0036	-0.0282	
100°C	-0.0096	-0.0463	

TABLE 6.3.2

## EVALUATION OF BASKET MANUFACTURING TOLERANCES

<b>Box I.D.</b>	<b>Box Wall Thickness</b>	<b>Maximum <math>k_{eff}</math></b>
MPC-37 (17x17B, 5.0% Enrichment)		
nominal (8.94")	nominal (0.59")	0.9332
nominal (8.94")	minimum (0.57")	0.9346
increased (8.96")	minimum (0.57")	0.9350
minimum (8.92")	minimum (0.57")	0.9352
minimum, including deformation (8.875")	minimum (0.57")	0.9374
MPC-89 (10x10A 4.8% Enrichment)		
nominal (6.01")	nominal (0.40")	0.9365
nominal (6.01")	minimum (0.38")	0.9403
increased (6.03")	minimum (0.38")	0.9396
minimum (5.99")	minimum (0.38")	0.9417
minimum, including deformation (5.96")	minimum (0.38")	0.9428

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TABLE 6.3.3

## BASKET DIMENSIONAL ASSUMPTIONS

<b>Basket Type</b>	<b>Box I.D.</b>	<b>Box Wall Thickness</b>
MPC-37	minimum, including deformation(8.875")	minimum (0.57")
MPC-89	minimum, including deformation (5.96")	minimum (0.38")

TABLE 6.3.4

## COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM FW SYSTEM

<b>UO<sub>2</sub>, DENSITY 10.686 g/cm<sup>3</sup> (97.5% of theoretical density of 10.96 g/cm<sup>3</sup>)</b>					
<b>Nuclide</b>	<b>Wgt. Fraction, 4.0 wt%</b>	<b>Wgt. Fraction, 4.5 wt%</b>	<b>Wgt. Fraction, 4.7 wt%</b>	<b>Wgt. Fraction, 4.8 wt%</b>	<b>Wgt. Fraction, 5.0 wt%</b>
8016	0.1185	0.1185	0.1185	0.1185	0.1185
92235	0.03526	0.03967	0.04143	0.04231	0.04408
92238	0.84624	0.84183	0.84007	0.83919	0.83742

<b>WATER (unborated and borated), DENSITY 1.0 g/cm<sup>3</sup></b>							
<b>Nuclide</b>	<b>Wgt. Fraction, 0 ppm</b>	<b>Wgt. Fraction, 1000 ppm</b>	<b>Wgt. Fraction, 1300 ppm</b>	<b>Wgt. Fraction, 1500 ppm</b>	<b>Wgt. Fraction, 1800 ppm</b>	<b>Wgt. Fraction, 2000 ppm</b>	<b>Wgt. Fraction, 2300 ppm</b>
5010	0.000E+00	1.800E-04	2.340E-04	2.700E-04	3.240E-04	3.600E-04	4.140E-04
5011	0.000E+00	8.200E-04	1.066E-03	1.230E-03	1.476E-03	1.640E-03	1.886E-03
1002	0.11190	0.11179	0.11175	0.11173	0.11170	0.11167	0.11164
8016	0.88810	0.88721	0.88695	0.88677	0.88650	0.88633	0.88606

<b>METAMIC HT, 9% B<sub>4</sub>C, DENSITY 2.6 g/cm<sup>3</sup></b>	
<b>Nuclide</b>	<b>Wgt. Fraction</b>
13027	0.91
6000	0.01956
5010	0.01289
5011	0.05755

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TABLE 6.3.4 (continued)

## COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM FW SYSTEM

<b>ZR CLAD, DENSITY 6.55 g/cm<sup>3</sup></b>	
<b>Nuclide</b>	<b>Wgt. Fraction</b>
40000	1.0
<b>STAINLESS STEEL, DENSITY 7.84 g/cm<sup>3</sup></b>	
<b>Nuclide</b>	<b>Wgt. Fraction</b>
24000	0.190
25055	0.020
26000	0.695
28000	0.095
<b>ALUMINUM, DENSITY 2.7 g/cm<sup>3</sup></b>	
<b>Nuclide</b>	<b>Wgt. Fraction</b>
13027	1.0
<b>CONCRETE, DENSITY 2.35 g/cm<sup>3</sup></b>	
<b>Nuclide</b>	<b>Wgt. Fraction</b>
1001	0.006
8016	0.500
11000	0.017
13027	0.048
14000	0.315
19000	0.019
20000	0.083
26000	0.012
<b>LEAD, DENSITY 11.34 g/cm<sup>3</sup></b>	
<b>Nuclide</b>	<b>Wgt. Fraction</b>
82000	1.0

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TABLE 6.3.5

REACTIVITY EFFECTS OF ECCENTRIC POSITIONING OF CONTENT  
(FUEL ASSEMBLIES AND DFCs) IN BASKET CELLS

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-37, Undamaged Fuel	0.9327	0.9380	0.0053	0.9143	-0.0184
MPC-37, Undamaged Fuel and Damaged Fuel/Fuel Debris (12 DFCs)	0.9260	0.9276	0.0016	0.9158	-0.0102
MPC-89, Undamaged Fuel	0.9369	0.9435	0.0066	0.9211	-0.0158
MPC-89, Undamaged Fuel and Damaged Fuel/Fuel Debris (16 DFCs)	0.9415	0.9451	0.0036	0.9301	-0.0114

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TABLE 6.3.6  
REACTIVITY EFFECTS GAPS IN BASKET CELL PLATES

Gaps in Metamic-HT	<b>MPC-37 (17x17B, 5.0% ENRICHMENT)</b>	<b>MPC-89 (10x10A, 4.8% ENRICHMENT)</b>
None	0.9380	0.9435
0.06" every 10"	0.9382	0.9439



TABLE 6.3.7

## RADIAL AND AXIAL DIMENSIONS OF THE HI-TRAC VW IN THE MCNP MODELS

<b>Component / Material</b>	<b>Thickness (Inches)</b>
<i>Radial Direction (Inside to Outside)</i>	
MPC Shell (Steel)	0.5
Water between MPC and HI-TRAC VW	0.125
HI-TRAC VW Shell (Steel)	0.75
HI-TRAC VW Lead	2.75
HI-TRAC VW Shell (Steel)	0.75
HI-TRAC VW Water Jacket	4.75
HI-TRAC VW Shell (Steel)	0.5
External Water Reflector	12
<i>Axial Direction (Bottom to Top)</i>	
External Water Reflector	12
HI-TRAC VW Bottom Lid (Steel)	5.5
MPC Base Plate (Steel)	3
Water	2
Active Fuel Region	150
Water	6
MPC Lid (Steel)	9
External Water Reflector	12

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TABLE 6.3.8

## MCNP CROSS SECTION SETS USED IN THE ANALYSES

Nuclide	MCNP Cross Section Set [6.1.4]
1001	62c
5010	66c
5011	66c
6000	66c
8016	62c
13027	62c
24000	50c
25055	62c
26000	55c
28000	50c
40000	66c
82000	50c
92235	69c
92238	69c

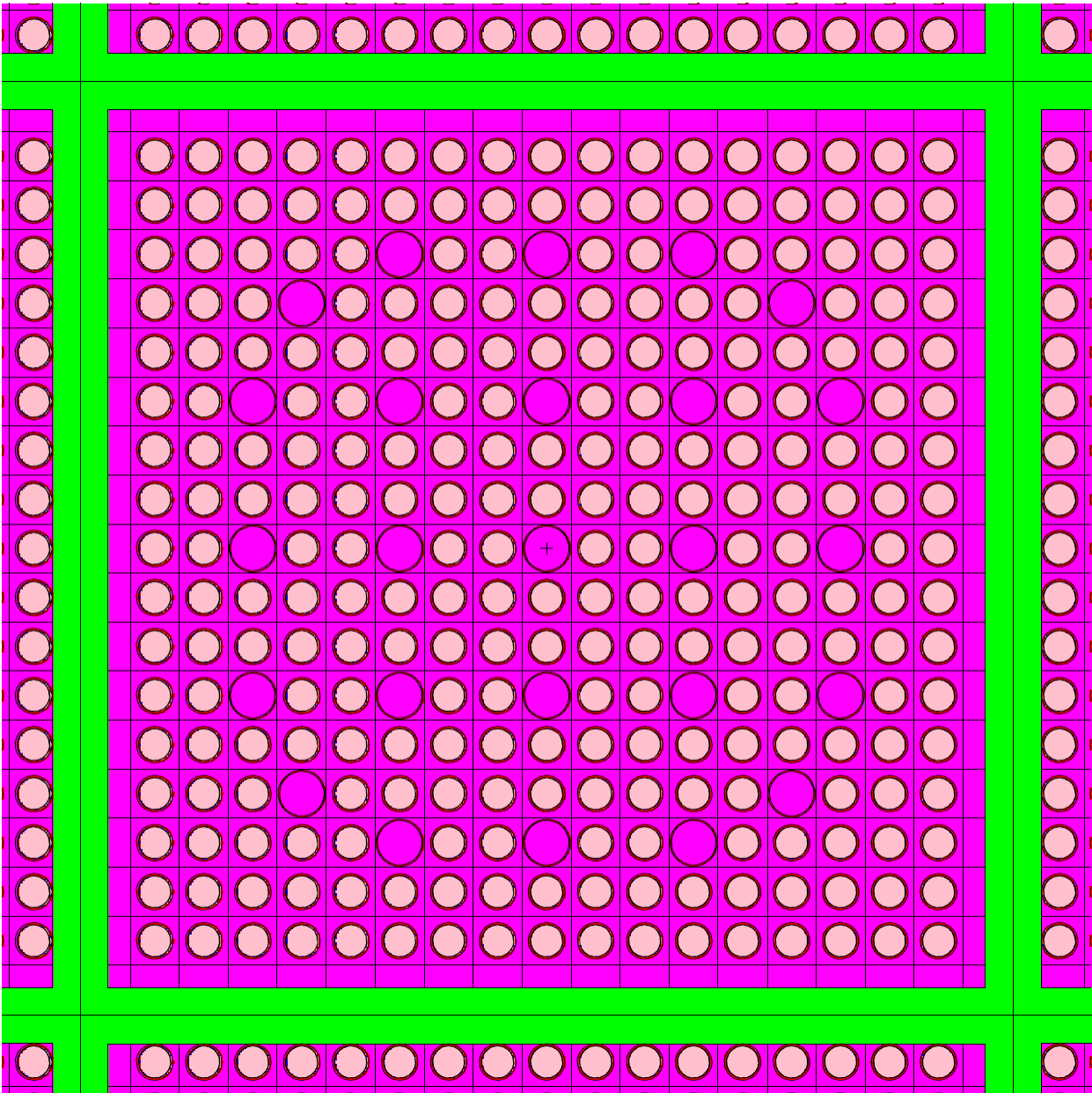


Figure generated directly from MCNP input file using the MCNP plot function. For Cell ID and Cell Wall Thickness see Table 6.3.3. For true dimensions see the drawings in Chapter 1.

Figure 6.3.1: Typical Cell of the Calculational Model (planar cross-section) with representative fuel in the MPC-37

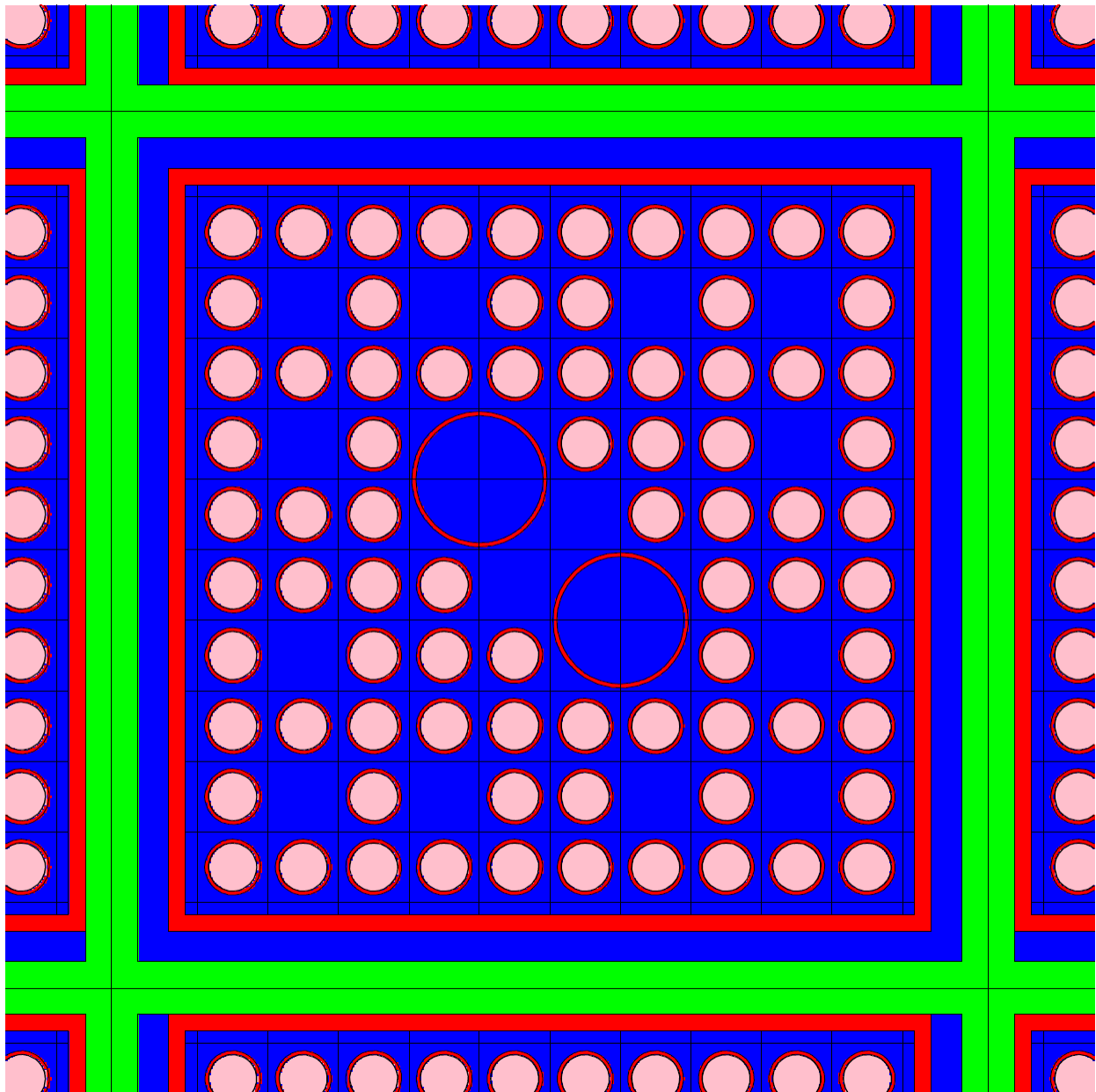


Figure generated directly from MCNP input file using the MCNP plot function. For Cell ID and Cell Wall Thickness see Table 6.3.3. For true dimensions see the drawings in Chapter 1.

Figure 6.3.2: Typical Cell of the Calculational Model (planar cross-section) with representative fuel in the MPC-89

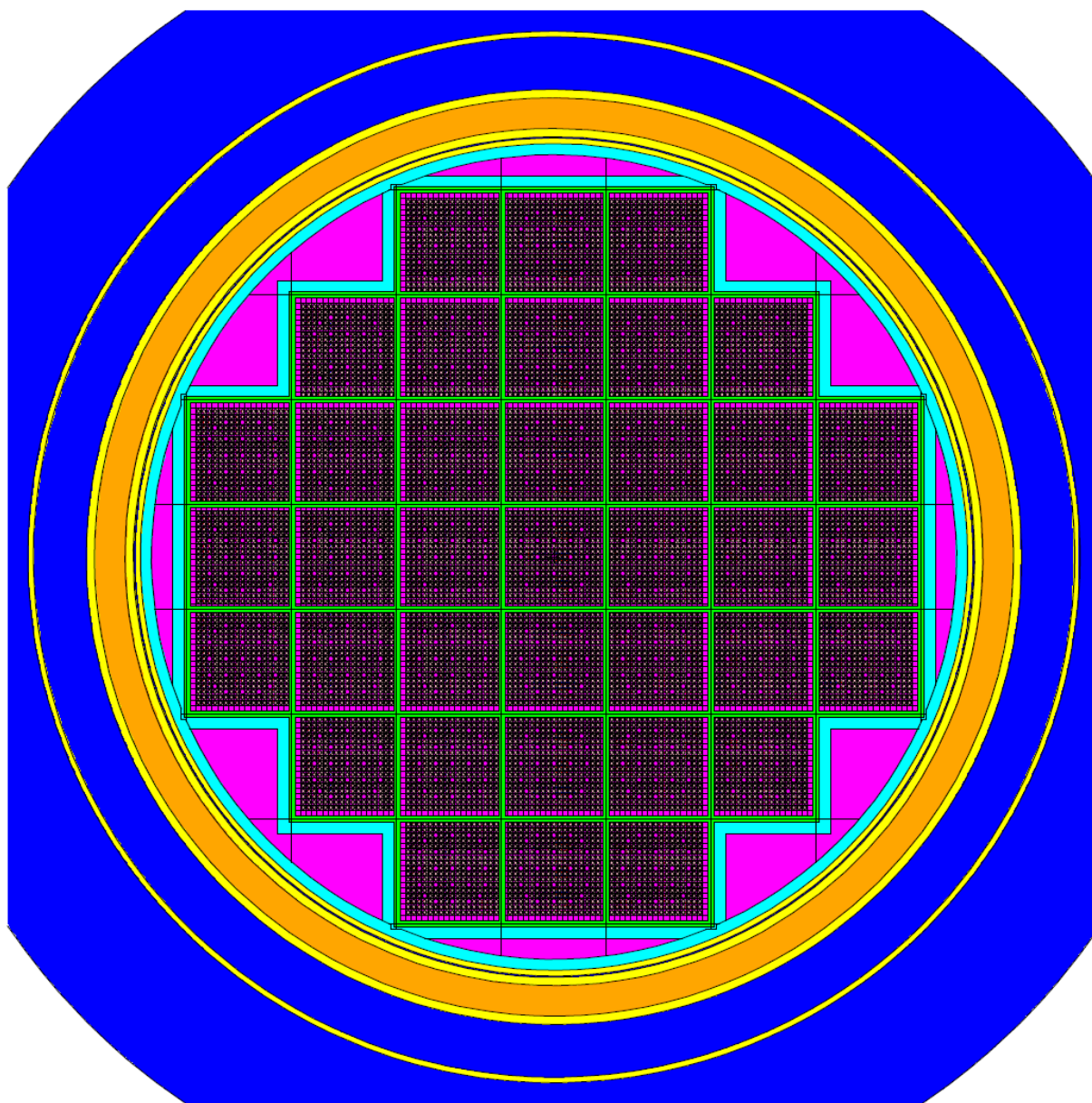


Figure generated directly from MCNP input file using the MCNP plot function. For radial dimensions of the HI-TRAC VW used in the analyses see Table 6.3.7. For true dimensions see the drawings in Chapter 1.

Figure 6.3.3: Calculational Model (planar cross-section) of the MPC-37

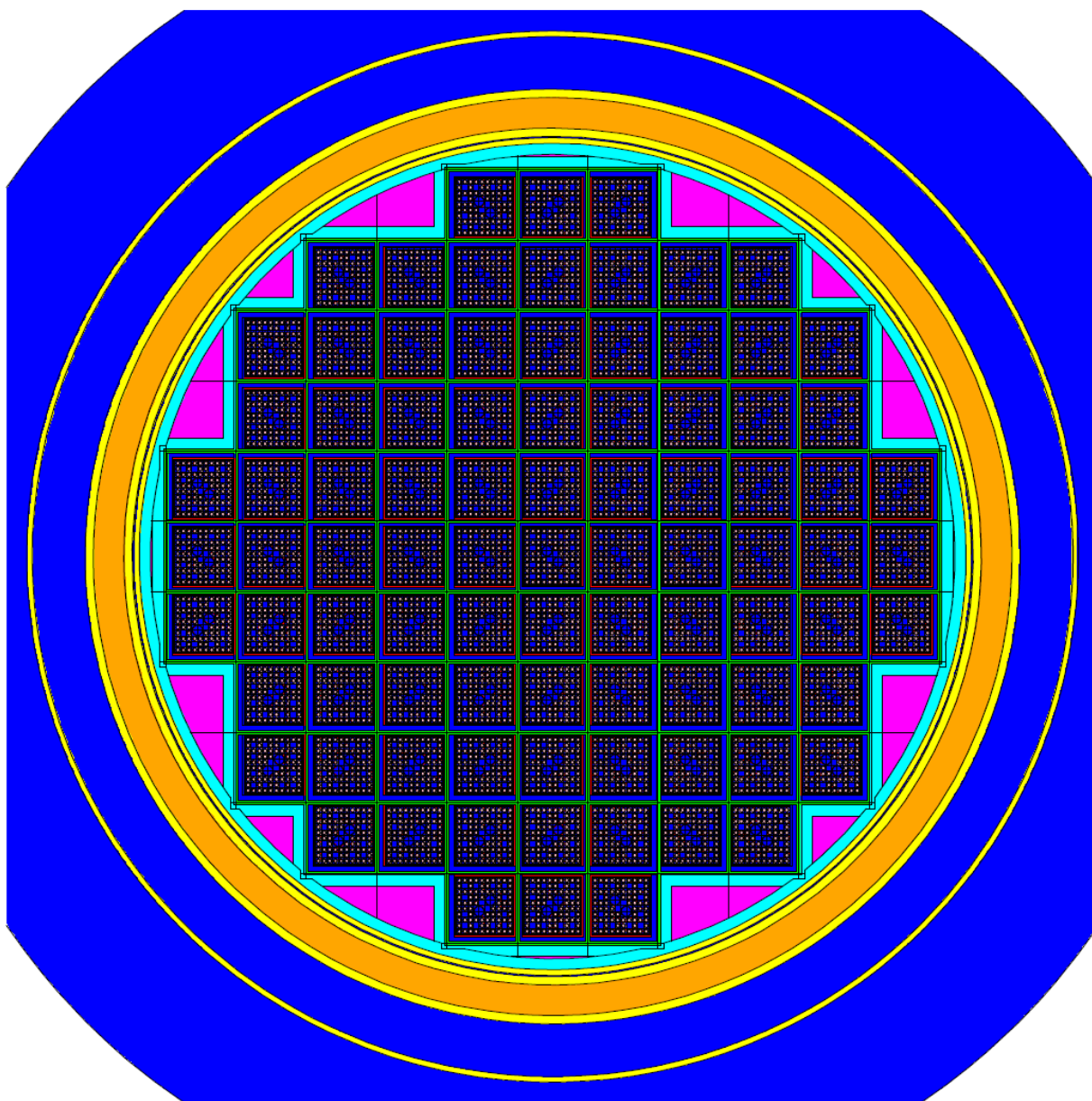


Figure generated directly from MCNP input file using the MCNP plot function. For radial dimensions of the HI-TRAC VW used in the analyses see Table 6.3.7. For true dimensions see the drawings in Chapter 1.

Figure 6.3.4: Calculational Model (planar cross-section) of the MPC-89



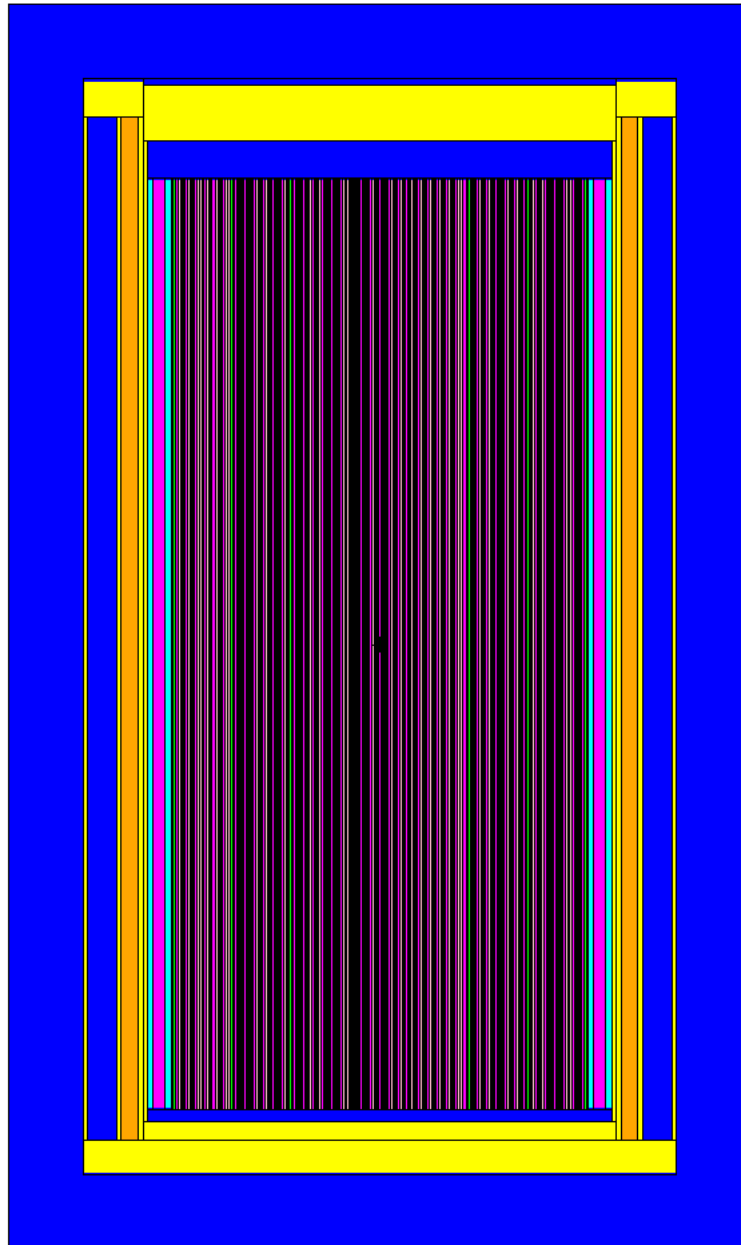


Figure generated directly from MCNP input file using the MCNP plot function. For axial dimensions of the HI-TRAC VW used in the analyses see Table 6.3.7. For true dimensions see the drawings in Chapter 1.

Figure 6.3.5: Calculational Model in Axial Direction

## 6.4 CRITICALITY CALCULATIONS

### 6.4.1 Calculational Methodology

The principal method for the criticality analysis is the general three-dimensional continuous energy Monte Carlo N-Particle code MCNP5 [6.1.4] developed at the Los Alamos National Laboratory. MCNP5 was selected because it has been extensively used and verified and has all of the necessary features for this analysis. MCNP5 calculations used continuous energy cross-section data distributed with the code [6.1.4].

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 MCNP5 criticality output contains a great deal of useful information that may be used to determine the acceptability of the problem convergence. Based on this information, a minimum of 20,000 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). To verify that these parameters are sufficient, studies were performed where the number of particles per cycle and/or the number of skipped cycles were increased. The calculations are presented in Table 6.4.9, and show only small differences between the cases, with the statistical tolerance of those calculations. All calculations are therefore performed with the parameters stated above, except for some studies that are performed with 50000 neutrons per cycle for improved accuracy, and except for the calculations for the HI-STORM, which need less particles for convergence. Appendix 6.D provides sample input files for the MPC-37 and MPC-89 basket in the HI-STORM FW system.

### 6.4.2 Fuel Loading or Other Contents Loading Optimization

The basket designs are intended to safely accommodate fuel with enrichments indicated in Section 2.1. The calculations were based on the assumption that the HI-STORM FW system (HI-TRAC VW transfer cask) was fully flooded with clean unborated water or water containing specific minimum soluble boron concentrations. 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.

The discussion provided in Section 6.2.1 regarding the principal characteristics of fuel assemblies and basket poison is also important for the various studies presented in this section, and supports the fact that those studies only need to be performed for a single BWR and PWR assembly types, and that the results of those studies are then generally applicable to all assembly types. The studies and the relationship to the discussion in Section 6.2.1 are listed below. Note that this approach is consistent with that used for the HI-STORM 100.



Internal and External Moderation (Section 6.4.2.1): The studies presented in Table 6.2.3 show that all assemblies essentially behave identical in respect to water moderation, specifically, that all assemblies are undermoderated. The principal effect of changes to the internal and external moderation would therefore be independent of the fuel type.

Partial Flooding (Section 6.4.2.2): The partial flooding of the basket, either in horizontal or vertical direction, reduces the amount of fuel that partakes effectively in the thermal fission process, while essentially maintaining the fuel-to-water ratio in the volume that is still flooded. This will therefore result in a reduction of the reactivity of the system (similar to that of the reduction of the active length), and due to the similarity of the fuel assemblies is not dependent on the specific fuel type.

Pellet-to-clad Gap (Section 6.4.2.3): As demonstrated by the studies shown in Section 6.2.1, all assemblies are undermoderated. Flooding the pellet-to-clad gap will therefore improve the moderation and therefore increase reactivity for all assembly types.

Preferential Flooding (Section 6.4.2.4): The only preferential flooding situation that may be credible is the flooding of the bottom section of the DFCs while the rest of the MPC internal cavity is already drained. In this condition, the undamaged assemblies have a negligible effect on the system reactivity since they are not flooded with water. The dominating effect is from the damaged fuel model in the DFCs. However, the damaged fuel model is conservatively based on an optimum moderated array of bare fuel rods in water, and therefore representative of all fuel types. The results are therefore applicable to all fuel types.

#### 6.4.2.1 Internal and External Moderation

Calculations in this section demonstrate that the HI-STORM FW system remains subcritical for all credible conditions of moderation.

##### 6.4.2.1.1 External Moderator Density

Calculations for the MPC designs with external moderators of various densities are shown in Table 6.4.1, all performed for the HI-TRAC VW and the MPC fully flooded. The results show that the maximum  $k_{\text{eff}}$  is essentially independent from the external water density. Nevertheless, all further evaluations are performed with full external water density.

##### 6.4.2.1.2 Internal Moderator Density

In a definitive study, Cano, et al. [6.4.1] have demonstrated that the phenomenon of a peak in reactivity at low moderator densities (sometimes called "optimum" moderation) does not occur in the presence of strong neutron absorbing material or in the absence of large water spaces between fuel assemblies in storage. Nevertheless, calculations were made to confirm that the phenomenon does not occur with low density water inside the casks.

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Calculations for the MPC designs with internal moderators of various densities are shown in Table 6.4.5. Results show that in all cases the reactivity reduces with reducing water density, with both filled and voided guide and instrument tubes for PWR assemblies (see Section 6.4.7). All further calculations are therefore performed with full water density inside the MPCs.

#### 6.4.2.2 Partial Flooding

Calculations in this section address partial flooding in the HI-STORM FW 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 ( $1.0 \text{ g/cm}^3$ ) water and the remainder of the cask is filled with steam consisting of ordinary water at a low partial density ( $0.002 \text{ g/cm}^3$  or less), 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 recommended 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  $k_{\text{eff}}$  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, the pellet-to-clad gap regions are assumed to be flooded with clean, unborated water.

#### 6.4.2.4 Preferential Flooding

Two different potential conditions of preferential flooding are considered: preferential flooding of the MPC basket itself (i.e., different water levels in different basket cells), and preferential flooding involving Damaged Fuel Containers.

Preferential flooding of the MPC basket itself for any of the MPC fuel basket designs is not possible because flow holes are present on all four walls of each basket cell at the bottom of the MPC basket. The flow holes are sized to ensure that they cannot be blocked by crud deposits (see Chapter 12). For damaged fuel assemblies and fuel debris, the assemblies or debris are loaded into stainless steel Damaged Fuel Containers fitted with mesh screens which prevent damaged fuel assemblies or fuel debris from blocking the basket flow holes. Preferential flooding of the MPC basket is therefore not possible.

However, when DFCs are present in the MPC, a condition could exist during the draining of the MPC, where the DFCs are still partly filled with water while the remainder of the MPC is dry. As a simplifying and conservative approach to model this condition it is assumed that the DFCs

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are completely flooded while the remainder of the MPC is only filled with steam consisting of ordinary water at partial density ( $0.002 \text{ g/cm}^3$  or less). Assuming this condition, the case resulting in the highest maximum  $k_{\text{eff}}$  for the fully flooded condition (see Subsection 6.4.4) is re-analyzed assuming the preferential flooding condition. Table 6.4.4 lists the maximum  $k_{\text{eff}}$  in comparison with the maximum  $k_{\text{eff}}$  for the fully flooded condition. For all configurations, the preferential flooding condition results in a lower maximum  $k_{\text{eff}}$  than the fully flooded condition. Thus, the preferential flooding condition is bounded by the fully flooded condition.

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. In summary, it is concluded that the MPC fuel baskets cannot be preferentially flooded, and that the potential preferential flooding conditions involving DFCs are bounded by the result for the fully flooded condition listed in Subsection 6.4.4.

#### 6.4.2.5 Design Basis Accidents

The analyses presented in Chapters 3 and 12 demonstrate that the damage resulting from the design basis accidents is limited to a loss of the water jacket for the HI-TRAC VW transfer cask and minor damage to the concrete radiation shield for the HI-STORM FW storage cask, which have no adverse effect on the design parameters important to criticality safety, and to minor deformation of the basket geometry, which is already considered in the analyses for the normal conditions.

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-STORM FW 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 design basis criticality safety calculations for the condition of full flooding with water (limiting cases) and summarized in Section 6.1. To demonstrate the applicability of the HI-TRAC VW analyses, results of the design basis criticality safety calculations for the HI-TRAC VW cask (limiting cases) are also summarized in Section 6.1 for comparison. These data confirm that for each of the candidate fuel types and basket configurations the effective multiplication factor ( $k_{\text{eff}}$ ), 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-4) 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.

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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 [6.1.5]. Each final  $k_{eff}$  value 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;
- ⇒  $\sigma_c$  is the standard deviation of the calculated  $k_{eff}$ , as determined by the computer code;
- ⇒ **Bias** is the systematic error in the calculations (code dependent) determined by comparison with critical experiments in Appendix 6.A; and
- ⇒  $\sigma_B$  is the standard error of the bias (which includes the K multiplier for 95% probability at the 95% confidence level; see Appendix 6.A).

The critical experiment benchmarking and the derivation of the bias and standard error of the bias (95% probability at the 95% confidence level) are presented in Appendix 6.A.

#### 6.4.4 Damaged Fuel and Fuel Debris

##### 6.4.4.1 Generic Approach

All MPCs are designed to contain PWR and BWR damaged fuel and fuel debris, loaded into DFCs. The number and permissible location of DFCs is provided in Table 2.1.1 and the licensing drawing in Section 1.5, respectively. Because the entire height of the fuel basket contains the neutron absorber (Metamic-HT), axial movement of DFC's does not have any reactivity consequence to MPC.

Damaged fuel assemblies, for the most part, are considered to be assemblies with known or suspected cladding defects greater than pinholes or hairline cracks, or with missing rods, but excluding fuel assemblies with gross defects (for the exact definition see the Glossary). Fuel debris can include a large variety of configurations ranging from whole fuel assemblies with severe damage down to individual fuel pellets.

To identify the configuration or configurations leading to the highest reactivity, a bounding approach is taken which is based on the analysis of regular arrays of bare fuel rods without cladding. Details and results of the analyses are discussed in the following subsections.

Note that since a modeling approach is used that bounds both damaged fuel and fuel debris without distinguishing between these two conditions, the term ‘damaged fuel’ as used throughout this chapter designates both damaged fuel and fuel debris.

Note that the modeling approach for damaged fuel and fuel debris is identical to that used in the HI-STORM 100 and HI-STAR 100.

### Bounding Undamaged Assemblies

The undamaged assemblies assumed in the basket in those cells not filled with DFCs are those that show the highest reactivity for each group of assemblies, namely

- 9x9E for BWR 9x9E/F, 8x8F and 10x10G assemblies
- 10x10F for BWR 10x10F assemblies
- 10x10A for all other BWR assemblies;
- 16x16A for all PWR assemblies with 14x14 and 16x16 arrays; and
- 15x15F for all PWR assemblies with 15x15 and 17x17 arrays.

Since the damaged fuel modeling approach results in higher reactivities, requirements of soluble boron for PWR fuel and maximum enrichment for BWR fuel are different from those for undamaged fuel only. Those limits are listed in Table 6.1.4 (PWR) and Table 6.1.5 (BWR) in Section 6.1. Note that for the calculational cases for damaged and undamaged fuel in the MPC-89, the same enrichment is used for the damage and undamaged assemblies.

Note that for the first group of BWR assemblies listed above (9x9E/F, 8x8F and 10x10G), calculations were performed for both 9x9E and 10x10G as undamaged assemblies, and assembly class 9x9E showed the higher reactivity, and is therefore used in the design basis analyses. This may seem contradictory to the results for undamaged assemblies listed in Table 6.1.2, where the 10x10G shows a higher reactivity. However, the cases in Table 6.1.2 are not at the same enrichment between those assemblies.

All calculations with damaged and undamaged fuel are performed for an active length of 150 inches. There are two assembly classes (17x17D and 17x17E) that have a larger active length for the undamaged fuel. However, the calculations for undamaged fuel presented in Table 6.1.1 show that the reactivity of those undamaged assemblies is at least 0.0050 delta-k lower than that of the assembly class 15x15F selected as the bounding assembly for the cases with undamaged and damaged fuel. The effect of the active fuel length is less than that, with a value of 0.0026 reported in Table 6.2.1 for a much larger difference in active length of 50 Inches. The difference in active length between the 17x17D/E and 15x15F is therefore more than bounded, and the 15x15F assembly class is therefore appropriate to bound all undamaged assemblies with 15x15 and 17x17 arrays.

### Bare Fuel Rod Arrays

A conservative approach is used to model both damaged fuel and fuel debris in the DFCs, using arrays of bare fuel rods:

- Fuel in the DFCs is arranged in regular, rectangular arrays of bare fuel rods, i.e., all cladding and other structural material in the DFC is replaced by water.
- For cases with soluble boron, additional calculations are performed with reduced water density in the DFC. This is to demonstrate that replacing all cladding and other structural material with borated water is conservative.
- The active length of these rods is assumed to be the same as for the intact fuel rods in the basket, even for more densely packed bare fuel rod arrays where it results in a total amount of fuel in the DFC that exceeds that for the intact assembly.
- To ensure the configuration with optimum moderation and highest reactivity is analyzed, the amount of fuel per unit length of the DFC is varied over a large range. This is achieved by changing the number of rods in the array and the rod pitch. The number of rods are varied between 16 (4x4) and 324 (18x18) for BWR fuel, and between 64 (8x8) and 576 (24x24) for PWR fuel.

This is a very conservative approach to model damaged fuel, and to model fuel debris configurations such as severely damaged assemblies and bundles of individual fuel rods, as the absorption in the cladding and structural material is neglected.

Further, this is a conservative approach to model fuel debris configurations such as bare fuel pellets due to the assumption of an active length of 150 inch (BWR and PWR). The actual height of bare fuel pellets in a DFC would be significantly below these values due to the limitation of the fuel mass for each basket position.

All calculations are performed for full cask models, containing the maximum permissible number of DFCs together with undamaged assemblies.

As an example of the damaged fuel model used in the analyses, Figure 6.4.1 shows the basket cell of an MPC-37 with a DFC containing a 14x14 array of bare fuel rods.

Principal results are listed in Table 6.4.6 and 6.4.7 for the MPC-37 and MPC-89, respectively. In all cases, the maximum  $k_{\text{eff}}$  is below the regulatory limit of 0.95.

For the HI-STORM 100, additional studies for damaged fuel assemblies were performed to further show that the above approach using arrays of bare fuel rods are bounding. The studies considered conditions including

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- Fuel assemblies that are undamaged except for various numbers of missing rods
- Variations in the diameter of the bare fuel rods in the arrays
- Consolidated fuel assemblies with clad rods
- Enrichment variations in BWR assemblies

Results of those studies were shown in the HI-STORM 100 FSAR, Table 6.4.8 and 6.4.9 and Figure 6.4.13 and 6.4.14 (undamaged and consolidated assemblies); HI-STORM 100 FSAR Table 6.4.12 and 6.4.13 (bare fuel rod diameter); and HI-STORM 100 FSAR Section 6.4.4.2.3 and Table 6.4.13 (enrichment variations). In all cases the results of those evaluations are equivalent to, or bounded by those for the bare fuel rods arrays. Since the generic approach of modeling damaged fuel and fuel debris is unchanged from the HI-STORM 100, these evaluations are still applicable and need not be re-performed for the HI-STORM FW.

#### 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 undamaged 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 Sealed Rods replacing BWR Water Rods

Some BWR fuel assemblies contain sealed rods filled with a non-fissile material 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.

#### 6.4.7 Non-fuel Hardware in PWR Fuel Assemblies

Non-fuel hardware as defined in Table 2.1.1 are permitted for storage with all PWR fuel types. Non-fuel hardware is inserted in the guide tubes of the assemblies, except for ITTRs, which are placed into the instrument tube.

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 and instrument tubes, i.e., any absorption of the hardware is neglected. Table 6.4.10 shows

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results for all PWR assembly classes at 5% enrichment with filled and voided guide and instrument tubes. These results show that for all classes, the condition with filled guide and instrument tubes bound those, or are statistically equivalent to those, with voided guide and instrument tubes. For the higher soluble boron concentration required in the presence of damaged fuel, the same is shown in Table 6.4.5 (two columns on the right). In this case, only the bounding case (Assembly class 15x15F as undamaged fuel) was analyzed.

In summary, 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.

#### 6.4.8 Neutron Sources in Fuel Assemblies

Fuel assemblies containing start-up neutron sources are permitted for storage in the HI-STORM FW 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 is true because in a system with a  $k_{\text{eff}}$  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.

#### 6.4.9 Low Enriched, Channeled BWR fuel

The calculations in this subsection show that low enriched, channeled BWR fuel with indeterminable cladding condition is acceptable for loading in all storage locations of the MPC-89 without placing those fuel assemblies into DFCs, hence classifying those assemblies as undamaged. The main characteristics that must be assured are:

- The channel is present and attached to the fuel assembly in the standard fashion; and
- The channel is essentially undamaged; and
- The maximum planar average enrichment of the assembly is less than or equal to 3.3 wt%  $^{235}\text{U}$

This analysis covers older assemblies, where the cladding integrity is uncertain, and where a verification of the cladding condition is prohibitive. An example of this type of fuel is the so-called CILC fuel, which has potential corrosion-induced damaged to the cladding but does not have grossly breached spent fuel rods.

The presence of the essentially undamaged and attached channel confines the fuel rods to a limited volume and the low enrichment, required for all assemblies in the MPC, limits the



reactivity of the fuel even under optimum moderation conditions. Due to the uncertain cladding condition, the analysis of this fuel follows essentially the same approach as that for the Damaged Fuel and Fuel Debris, i.e. bare fuel rod arrays of varying sizes are analyzed within the confines of the channel. This is an extremely conservative modeling approach for this condition, since reconfiguration is not expected and cladding would still be present. The results of this conservative analysis are listed in Table 6.4.8 and show that the system remains below the regulatory limit with these assemblies in all cells of the MPC-89, without DFCs.

These results confirm that even with unknown cladding condition the maximum  $k_{\text{eff}}$  values are below the regulatory limit when fully flooded and loaded with any of the BWR candidate fuel assemblies, therefore if the cladding is not grossly breached and the fuel assembly structurally sound it can be considered undamaged when loading in an MPC-89.

TABLE 6.4.1

## MAXIMUM REACTIVITIES WITH REDUCED EXTERNAL WATER DENSITIES

Water Density		Maximum $k_{eff}$	
Internal	External	MPC-37 (17x17B, 5.0%)	MPC-89 (10x10A, 4.8%)
100%	100%	0.9380	0.9435
100%	70%	0.9377	0.9432
100%	50%	0.9399	0.9439
100%	20%	0.9366	0.9428
100%	10%	0.9374	0.9437
100%	5%	0.9376	0.9435
100%	1%	0.9383	0.9435

TABLE 6.4.2

## REACTIVITY EFFECTS OF PARTIAL CASK FLOODING

<b>MPC-37 (17x17B, 5.0% ENRICHMENT)</b>		
Flooded Condition (% Full)	Maximum $k_{\text{eff}}$ , Vertical Orientation	Maximum $k_{\text{eff}}$ , Horizontal Orientation
25	0.9175	0.8306
50	0.9325	0.9093
75	0.9357	0.9349
100	0.9380	0.9380
<b>MPC-89 (10x10A, 4.8% ENRICHMENT)</b>		
Flooded Condition (% Full)	Maximum $k_{\text{eff}}$ , Vertical Orientation	Maximum $k_{\text{eff}}$ , Horizontal Orientation
25	0.9204	0.8345
50	0.9382	0.9128
75	0.9416	0.9392
100	0.9435	0.9435

TABLE 6.4.3

REACTIVITY EFFECT OF FLOODING THE  
PELLET-TO-CLAD GAP

Pellet-to-Clad Condition	<b>Maximum <math>k_{eff}</math></b>	
	<b>MPC-37 (17x17B, 5.0% ENRICHMENT)</b>	<b>MPC-89 (10x10A, 4.8% ENRICHMENT)</b>
dry	0.9335	0.9391
flooded with unborated water	0.9380	0.9435

TABLE 6.4.4

## REACTIVITY EFFECT OF PREFERENTIAL FLOODING OF THE DFCs

DFC Configuration	Maximum $k_{eff}$	
	Preferential Flooding	Fully Flooded
MPC-37 with 12 DFCs (5% Enrichment, Undamaged assembly 15x15F, 20x20 Bare Rod Array)	0.8705	0.9276
MPC-89 with 16 DFCs (4.8 % Enrichment, Undamaged assembly 10x10A, 9x9 Bare Rod Array)	0.8296	0.9464

TABLE 6.4.5

MAXIMUM  $k_{eff}$  VALUES WITH REDUCED  
WATER DENSITIES

Internal Water Density <sup>†</sup> in g/cm <sup>3</sup>	Maximum $k_{eff}$						
	MPC-89 10x10A, 4.8%	MPC-37 (1500ppm) 17x17B, 4.0 %		MPC-37 (2000ppm) 17x17B, 5.0 %		MPC-37 <sup>†</sup> (2300ppm) 15x15F and Damaged Fuel 5.0 %	
Guide Tubes	N/A	filled	void	filled	void	filled	void
1.00	0.9435	0.9181	0.9071	0.9380	0.9292	0.9276	0.9265
0.99	0.9415	0.9181	0.9059	0.9367	0.9296	0.9271	0.9264
0.98	0.9391	0.9162	0.9054	0.9368	0.9279	0.9271	0.9257
0.97	0.9370	0.9166	0.9035	0.9364	0.9272	0.9265	0.9242
0.96	0.9345	0.9147	0.9005	0.9360	0.9265	0.9265	0.9232
0.95	0.9304	0.9148	0.9010	0.9356	0.9243	0.9253	0.9217
0.94	0.9280	0.9133	0.8995	0.9335	0.9238	0.9255	0.9225
0.93	0.9259	0.9128	0.8986	0.9355	0.9237	0.9263	0.9214
0.92	0.9232	0.9120	0.8955	0.9327	0.9203	0.9237	0.9204
0.91	0.9183	0.9105	0.8947	0.9335	0.9208	0.9229	0.9194
0.90	0.9169	0.9090	0.8934	0.9303	0.9189	0.9226	0.9169
0.85	0.9013	0.9042	0.8840	0.9272	0.9109	0.9190	0.9127
0.80	0.8850	0.8973	0.8733	0.9222	0.9022	0.9138	0.9040
0.70	0.8462	0.8813	0.8477	0.9068	0.8780	0.9000	0.8851
0.60	0.7980	0.8565	0.8132	0.8866	0.8478	0.8806	0.8571
0.40	0.6762	0.7876	0.7195	0.8244	0.7585	0.8192	0.7735
0.20	0.5268	0.6827	0.5806	0.7284	0.6298	0.7237	0.6517
0.10	0.4649	0.6206	0.5112	0.6698	0.5639	0.6669	0.5889

<sup>†</sup> External moderator is modeled at 100%.

<sup>†</sup> With undamaged and damaged fuel. All other cases with undamaged fuel only

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TABLE 6.4.6

MAXIMUM  $k_{\text{eff}}$  VALUES IN THE MPC-37 WITH UNDAMAGED (15x15F)  
AND DAMAGED FUEL

<b>Bare Rod Array inside the DFC</b>	<b>Maximum <math>k_{\text{eff}}</math>, 4.0 wt%</b>	<b>Maximum <math>k_{\text{eff}}</math>, 5.0 wt%</b>
8x8	0.8883	0.9122
10x10	0.8899	0.9135
12x12	0.8910	0.9152
14x14	0.8945	0.9177
15x15	0.8966	0.9198
16x16	0.8982	0.9224
17x17	0.9003	0.9238
18x18	0.9027	0.9262
20x20	0.9032	0.9276
22x22	0.9023	0.926
24x24	0.9008	0.9239

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TABLE 6.4.7

MAXIMUM  $k_{\text{eff}}$  VALUES IN THE MPC-89 WITH UNDAMAGED (10x10A)  
AND DAMAGED FUEL

<b>Bare Rod Array inside the DFC</b>	<b>Maximum <math>k_{\text{eff}}</math>, 4.8 wt% (planar average)</b>
4x4	0.9389
6x6	0.9411
8x8	0.9432
9x9	0.9464
10x10	0.9454
11x11	0.9451
12x12	0.9460
13x13	0.9453
14x14	0.9444
16x16	0.9429
18x18	0.9423



TABLE 6.4.8  
 MAXIMUM  $k_{\text{eff}}$  VALUES IN THE MPC-89 WITH LOW ENRICHED (3.3  
 wt%  $^{235}\text{U}$ ), CHANNELED, BWR FUEL

<b>Rod Array inside the Channel</b>	<b>Maximum <math>k_{\text{eff}}</math></b>
4x4	0.4018
6x6	0.7320
8x8	0.8999
9x9	0.9294
10x10	0.9325
11x11	0.9131
12x12	0.8762
13x13	0.8237
14x14	0.7606
16x16	0.6664
18x18	0.6334

TABLE 6.4.9

## COMPARISON OF MCNP CONVERGENCE PARAMETERS

Calculation Parameters		Maximum $k_{\text{eff}}$	
Particles per Cycle	Skipped Cycles	<b>MPC-37 (17x17B, 5.0% ENRICHMENT)</b>	<b>MPC-89 (10x10A, 4.8% ENRICHMENT)</b>
20,000	20	0.9380	0.9435
50,000	20	0.9376	0.9428
20,000	100	0.9387	0.9436
50,000	100	0.9379	0.9434

TABLE 6.4.10

COMPARISON OF MAXIMUM  $k_{\text{eff}}$  VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-37 WITH CONDITIONS OF FILLED AND VOIDED GUIDE AND INSTRUMENT TUBES AT 5 % ENRICHMENT

<b>Fuel Assembly Class</b>	<b>Maximum <math>k_{\text{eff}}</math>, Filled Tubes</b>	<b>Maximum <math>k_{\text{eff}}</math>, Voided Tubes</b>
14x14A	0.8983	0.8887
14x14B	0.9282	0.9148
14x14C	0.9275	0.9277
15x15B	0.9311	0.9251
15x15C	0.9188	0.9134
15x15D	0.9421	0.9379
15x15E	0.9410	0.9365
15x15F	0.9455	0.9404
15x15H	0.9325	0.9317
15x15I	0.9357	0.9362
16x16A	0.9366	0.9320
17x17A	0.9194	0.9135
17x17B	0.9380	0.9292
17x17C	0.9424	0.9345
17x17D	0.9384	0.9293
17x17E	0.9392	0.9314

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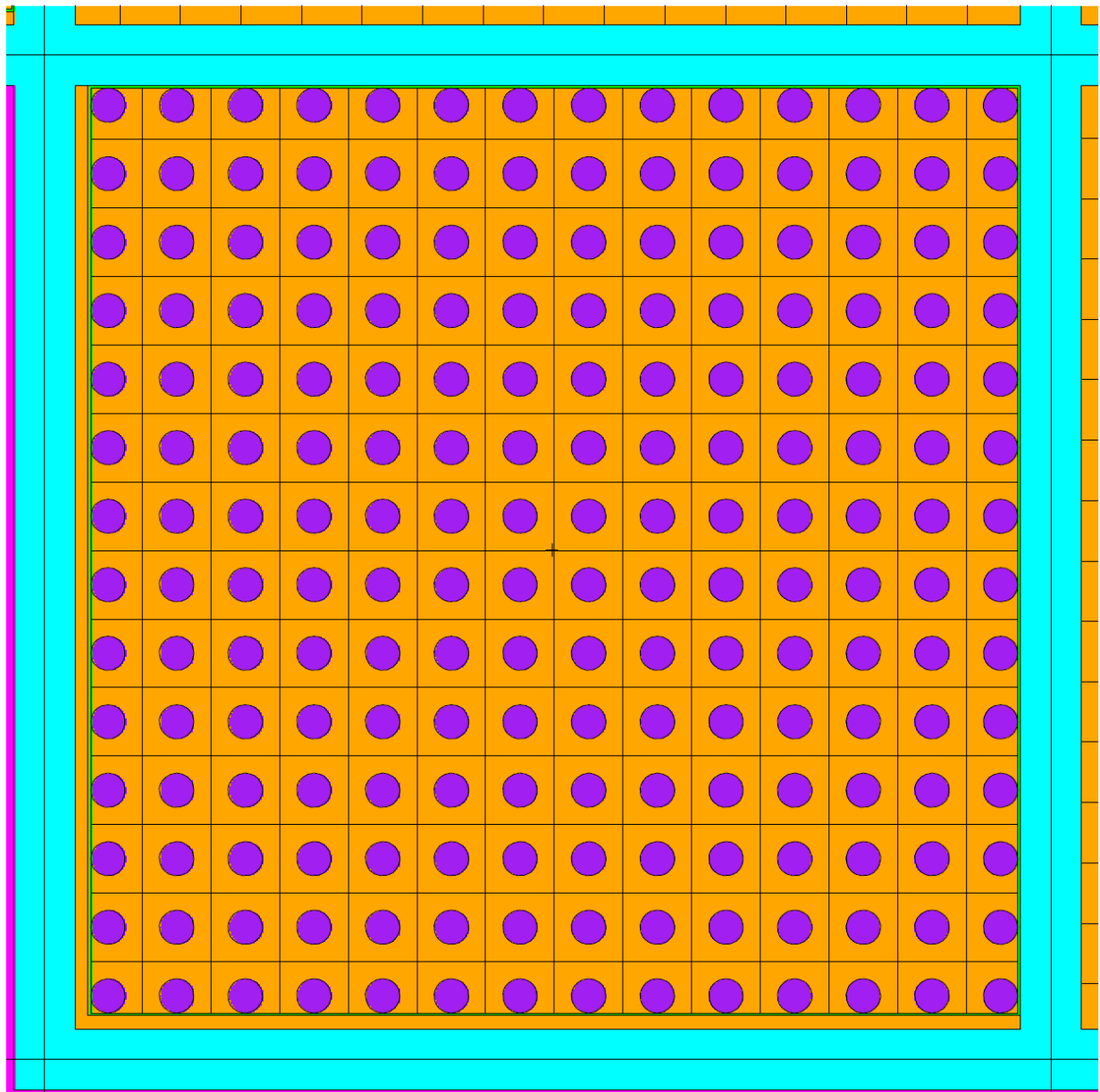


Figure 6.4.1: Calculational Model (planar cross-section) of a DFC in a MPC-37 cell with a 14x14 array of bare fuel rods

## 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) cell spacing, and (3) the  $^{10}\text{B}$  loading of the neutron absorber panels. 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  $k_{\text{eff}}$  values for the cask. Further, all calculations were performed on the same computer hardware (personal computers).

## 6.6 REGULATORY COMPLIANCE

This section documents the criticality evaluation of the HI-STORM FW system for the storage of spent nuclear fuel. This evaluation demonstrates that the HI-STORM FW system is in full compliance with the criticality requirements of 10CFR72 and NUREG-1536.

Structures, systems, and components important to criticality safety, as well as the limiting fuel characteristics, are described in sufficient detail in this section to enable an evaluation of their effectiveness.

The HI-STORM FW 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 poison has shown that they will remain effective for a storage period greater than 60 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 60 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-STORM FW system is in full compliance with the double contingency requirements of 10CFR72.124. Therefore, it is concluded that the criticality design features for the HI-STORM FW 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-STORM FW system will allow safe storage of spent fuel.

## 6.7 REFERENCES

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## **APPENDIX 6.A: BENCHMARK CALCULATIONS**

**Withheld in Accordance with 10 CFR 2.390**



## **APPENDIX 6.B: MISCELLANEOUS INFORMATION**

- 6.B.1 Sample Input File MPC-37
- 6.B.2 Sample Input File MPC-89
- 6.B.3 Analyzed Distributed Enrichment Patterns for Higher Enrichments
- 6.B.4 Assembly Cross Sections

6.B.1 Sample Input File MPC-37

**Withheld in Accordance with 10 CFR 2.390**

6.B.2 Sample Input File MPC-89

**Withheld in Accordance with 10 CFR 2.390**

6.B.3 Analyzed Distributed Enrichment Patterns

**Withheld in Accordance with 10 CFR 2.390**

6.B.4 Assembly Cross Sections

**Withheld in Accordance with 10 CFR 2.390**

## CHAPTER 7\*: CONFINEMENT

### 7.0 INTRODUCTION

Confinement of all radioactive materials in the HI-STORM FW system is provided by the MPC. The design of the HI-STORM FW MPC assures that there are no credible design basis events that would result in a radiological release to the environment. The HI-STORM FW overpack and HI-TRAC VW 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-STORM FW 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-STORM FW 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-STORM FW 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.

\*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.

## 7.1 CONFINEMENT DESIGN CHARACTERISTICS

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 base 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-STORM FW 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 10.1.1.

#### 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, providing 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.

#### 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.

<p style="text-align: center;">Table 7.1.1</p> <p style="text-align: center;">MPC CONFINEMENT BOUNDARY WELDS</p>		
<b>MPC Weld Location</b>	<b>Weld Type<sup>†</sup></b>	<b>ASME Code Category (Section III, Subsection NB)</b>
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 covers to MPC lid	Partial Penetration Groove (field weld)	D
MPC closure ring to MPC lid	Partial Penetration Groove (field weld)	C
MPC closure ring to closure ring radial weld	Partial Penetration Groove (field weld)	††

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<sup>†</sup> The tests and inspections for the Confinement Boundary welds are listed in Section 10.1

<sup>††</sup> This joint is governed by NB-5271 (liquid penetrant examination).

Table 7.1.2

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-STORM FW 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 REQUIREMENTS FOR NORMAL AND OFF-NORMAL CONDITIONS OF STORAGE

Once sealed and transferred into the HI-STORM FW 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 an explosive level of gases due to the radiological decomposition in the MPC is eliminated by excluding foreign materials in the MPC or by evaluating foreign material to demonstrate the effect on the MPC internal pressure is negligible.

### 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.

## 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.
- [7.1.1] 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: MATERIAL EVALUATION

### 8.1 INTRODUCTION

This chapter presents an assessment of the materials selected for use in the HI-STORM FW System components identified in the licensing drawings in Section 1.5. In this chapter and Chapter 3 of this FSAR, the significant mechanical, thermal, radiological and metallurgical properties of materials identified for use in the components of the HI-STORM FW System are presented. This chapter focuses on the HI-STORM FW material properties to assess compliance with the ISG-15 [8.1.1] and ISG-11 [8.1.2] requirements. The principal purpose of ISG-15 is to evaluate the dry cask storage system to ensure adequate material performance of the independent spent fuel storage installation (ISFSI) components designated as important to safety under normal, off-normal and accident conditions. Some areas of review applicable to the suitability assessment of the materials have been addressed elsewhere in this FSAR and are referenced from this chapter as necessary. Areas that require further details are reviewed within this chapter as necessary to satisfy the requirements of ISG-15. Guidance on performing the review is adopted directly from ISG-15 and ISG-11.

ISG-15 sets down the following general acceptance criteria for material evaluation.

- The safety analysis report should describe all materials used for dry spent fuel storage components designated as important to safety, and should consider the suitability of those materials for their intended functions in sufficient detail to evaluate their effectiveness in relation to all safety functions.
- The dry spent fuel storage system should employ materials that are compatible with wet and dry spent fuel loading and unloading operations and facilities. These materials should not degrade to the extent that a safety concern is created.

The information compiled in this chapter addresses the above acceptance criteria. To perform the material suitability evaluation, it is necessary to characterize the following for each component: (i) the applicable environment, (ii) the potential degradation modes and (iii) the potential hazards to continued effectiveness of the selected material.

The operating environments of the different components of the cask system are not the same. Likewise, the potential degradation modes and hazards are different for each component. Tables 8.1.1, 8.1.2, and 8.1.3 provide a summary of the environmental states, potential degradation modes and hazards applicable to the MPC, the HI-STORM FW overpack and the HI-TRAC VW cask, respectively. The above referenced tables serve to guide the material suitability evaluation for the HI-STORM FW System.

To provide a proper context for the subsequent evaluations, the potential degradation mechanisms applicable to the ventilated systems are summarized in Table 8.1.4. The degradation mechanisms listed in Table 8.1.4 are considered in the suitability evaluation presented later in this chapter.

The material evaluation presented in this chapter is intended to be complete, even though a conclusion of the adequacy of the materials can be made on the strength of the following facts:

- i. The materials used in HI-STORM FW are, with the sole exception of Metamic-HT, identical to those used in the widely deployed HI-STORM 100 System (Docket No. 72-1014).
- ii. The thermal environment in the HI-STORM FW system emulates the HI-STORM 100 system in all respects.
- iii. The fuel loading and short-term operations are essentially identical to those that have been practiced in the HI-STORM 100 system throughout the industry.

Table 8.1.1

## CONSIDERATIONS GERMANE TO THE MPC MATERIAL PERFORMANCE

<b>Consideration</b>	<b>Short-Term Operations</b>	<b>Long-Term Storage</b>
Environment	Aqueous (with Boric acid in PWR plants), characterized by moderately hot (<212°F) water during fuel loading, rapid evaporation during welding and drying operations	MPC's internal environment is hot ( $\leq 752^{\circ}\text{F}$ ), inertized and dry. Temperature of the MPC cycles very gradually due to changes in the environmental temperature.
Potential degradation modes	Hydrogen generation from oxidation of aluminum and aluminum alloy internals. Risk to the integrity of fuel cladding from thermal transients caused by vacuum drying.	Corrosion of the external surfaces of the MPC (stress, corrosion, cracking, pitting, etc.)
Potential hazards to effective performance	Inadequate drying of waterlogged fuel rods.	Blockage of ventilation ducts under an extreme environmental phenomenon leading to a rapid heat-up of the MPC internals.

Table 8.1.2

**CONSIDERATIONS GERMANE TO THE HI-STORM FW OVERPACK  
MATERIAL PERFORMANCE\***

<b>Consideration</b>	<b>Performance Data</b>
Environment	Cool ambient air progressively heated as it rises in the overpack/MPC annulus heating the inside surface of the cask. The heated air has reduced relative humidity. Direct heating of the overpack inner shell by radiation can be prevented using the optional “heat shields” described in Chapter 1, on a site specific basis. External surface of the overpack including the top lid is heated and in contact with ambient air, rain, and snow, as applicable.
Potential degradation modes	Peeling or perforation of surface preservatives and corrosion of any exposed steel surfaces.
Potential hazards to effective performance	Blockage of ducts by debris leading to overheating of the concrete in the overpack, scorching of the cask by proximate fire, lightning.

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\* Short-term operations are not applicable to the HI-STORM FW overpack.

Table 8.1.3 CONSIDERATIONS GERMANE TO THE OF HI-TRAC VW MATERIAL PERFORMANCE*	
Consideration	Performance Data
Environment	Heated fuel pool water on the outside and demineralized water in contact with the inside surface, heated water in the “water jacket”. Temperature ramps on the inside surface during the drying and “backfill” operation.
Potential degradation modes	Peeling or perforation of surface coatings, loss of effectiveness of bottom lid gasket.
Potential hazards to effective performance	Lead slump due to sudden inertial loading; contamination of the inside surface of the cask by pool water, partial loss of heat rejection resulting in boiling of water in the water jacket, impact from tornado missile during transfer to the ISFSI.

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\* Long-term storage conditions are not applicable to the transfer cask.



Table 8.1.4 FAILURE AND DEGRADATION MECHANISMS APPLICABLE TO VENTILATED SYSTEMS <sup>§</sup>			
	<b>Mechanism</b>	<b>Area of Performance Affected</b>	<b>Vulnerable Parts</b>
1.	General Corrosion	Structural capacity	All carbon steel parts
2.	Hydrogen Generation	Personnel safety during short-term operations	Coatings, parts made of aluminum or aluminum alloys
3.	Stress Corrosion Cracking	Structural	Austenitic Stainless Steel
4.	Creep	Criticality control	Fuel Basket
5.	Galling	Equipment handling and deployment	Threaded Fasteners
6.	Hysteresis	During fuel loading in the pool	HI-TRAC VW Bottom Lid Gaskets
7.	Fatigue	Structural Integrity	Fuel Cladding & Bolting
8.	Brittle Fracture	Structural Capacity	Thick Steel Parts
9.	Boron Depletion	Criticality Control	Neutron Absorber

<sup>§</sup> This table lists all potential (generic) mechanisms, whether they are credible for the HI-STORM FW System or not. The viability of each failure mechanism is discussed later in this chapter.

## 8.2 MATERIAL SELECTION

The following acceptance criteria are applicable for material selection per ISG-15.

- a. The material properties of a dry spent fuel storage component should meet its service requirements in the proposed cask system for the duration of the licensing period.
- b. The materials that comprise the dry spent fuel storage should maintain their physical and mechanical properties during all conditions of operations. The spent fuel should be readily retrievable without posing operational safety problems.
- c. Over the range of temperatures expected prior to and during the storage period, any ductile-to-brittle transition of the dry spent fuel storage materials, used for structural and nonstructural components, should be evaluated for its effects on safety.
- d. Dry spent fuel storage gamma shielding materials (e.g. lead) should not experience slumping or loss of shielding effectiveness to an extent that compromises safety. The shield should perform its intended function throughout the licensed service period.
- e. Dry spent fuel storage materials used for neutron absorption should be designed to perform their safety function.
- f. Dry spent fuel storage protective coatings should remain intact and adherent during all loading and unloading operations within wet or dry spent fuel facilities, and during long-term storage.

The above criteria have been utilized in selecting the material types for the HI-STORM FW system. The selected materials provide the required heat transfer, confinement, shielding and the criticality control of the stored spent fuel and are capable of withstanding loadings including seismic, temperature cycles due to internal heat and ambient temperature variation, extreme temperature conditions, loads due to natural phenomena like tornado missiles, flooding and other credible hypothetical accident scenarios. The HI-STORM FW components must withstand the environmental conditions experienced during normal operation, off-normal conditions and accident conditions for the entire service life.

The selection of materials is guided by the applicable loadings and potential failure modes. An emphasis has been placed on utilizing proven materials that have established properties and characteristics and are of proven reliability. Where a relatively new material (e.g., Metamic-HT) is used, comprehensive tests have been conducted to ensure reliability.

The major structural materials used in HI-STORM FW System are discussed in this section. The mechanical and thermal properties of these materials are presented in Section 8.4. The materials for welds are discussed in Section 8.5. The structural materials for bolts and fasteners are discussed in Section 8.6. Coatings and paints are discussed in Section 8.7. Gamma and neutron

shielding materials are treated in Section 8.8. The neutron absorbing materials are discussed in Section 8.9.

Chapter 1 provides a general description of the HI-STORM FW System including information on materials of construction. All materials of construction are identified in the drawing package provided in Section 1.5 and the ITS categories of the sub-components are identified in Table 2.0.1 through 2.0.8.

## 8.2.1 Structural Materials

### 8.2.1.1 Cask Components and Their Constituent Materials

The major structural materials that are used in the HI-STORM FW System are Alloy X, Metamic-HT, carbon steel, and aluminum. They are further discussed below in light of the ISG-15 requirements.

#### MPC

All structural components in an MPC Enclosure Vessel are made of Alloy X (stainless steel). Appendix 1.A provides discussions on Alloy X materials. The fuel basket is made of Metamic-HT neutron absorber described in Chapter 1, Section 1.2.1.4. The confinement boundary is made of stainless steel material for its superior strength, ductility, and resistance to corrosion and brittle fracture for long term storage. The basket shims used to support the basket are made of a creep resistant aluminum alloy. The two-piece MPC lid is either made entirely of Alloy X or the bottom portion of the lid is made of carbon steel with stainless steel veneer. The principal materials used in the fabrication of the MPC are listed in Section 1.2.

#### HI-STORM

The main structural function of the overpack is provided by carbon steel and the main shielding function is provided by plain concrete. Chapter 1 presents discussions on these materials. The materials used in the fabrication of the overpack are listed in Section 1.2.

#### HI-TRAC

As discussed in Chapter 1, the HI-TRAC VW transfer cask is principally made of carbon steel and lead. The HI-TRAC VW is equipped with a water jacket. The materials used in the fabrication of the transfer cask are listed in Section 1.2.

### 8.2.1.2 Synopsis of Structural Materials

#### i. Alloy X

The MPC enclosure vessel design allows use of any one of the four Alloy X materials: Types 304, 304LN, 316 and 316LN. 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. Each of these material properties are provided in the ASME Code Section II [8.3.1].

As discussed in Appendix 1.A, the Alloy X approach is conservative because, no matter which material is ultimately utilized, the Alloy X guarantees that the performance of the MPC will meet or exceed the analytical predictions. The material properties are provided at various temperatures.

All structural analyses utilize conservatively established material properties such as design stress intensity, tensile strength, yield strength, and coefficient of thermal expansion for the range of temperature conditions that would be experienced by the cask components.

Chapter 3 provides the structural evaluation for the MPC Enclosure Vessel which is made of Alloy X. It is demonstrated that Alloy X provides adequate structural integrity for the MPC enclosure vessel under normal, off normal, and accident conditions. As shown in Chapter 4, the maximum metal temperature for Alloy X for the Confinement Boundary remains the design temperatures in Table 2.2.3 under all service modes. 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 1000°F temperature for up to 10,000 hours.

Since stainless steel materials do not undergo a ductile-to-brittle transition in the minimum permissible service temperature range of the HI-STORM FW System, brittle fracture is not a concern for the MPC components. Subsection 8.4.3 presents further discussions on brittle fracture.

In Section 8.12, the potential for chemical and galvanic reaction of Alloy X in short-term and long-term operating conditions is evaluated. Alloy X is also used in the Confinement Boundary of all HI-STORM 100 MPCs.

## ii. Metamic-HT

Criticality control in the HI-STORM FW System is provided by the coplanar grid work of the Fuel Basket honeycomb, made entirely of the Metamic-HT extruded metal matrix composite plates. The boron in Metamic-HT provides criticality control in the HI-STORM FW System. The Metamic-HT neutron absorber is a successor to the Metamic (classic) product widely used in dry storage fuel baskets and spent fuel storage racks (the “HT” designation in Metamic-HT stands for high temperature and is derived from this characteristic). Metamic-HT has been licensed in the HI-STAR 180 transport cask (Docket No. 71-9325).

Metamic-HT is also engineered to possess the necessary mechanical characteristics for structural application. The mechanical properties of Metamic-HT are derived from the strengthening of its aluminum matrix with ultra fine-grained (nano-particle size) alumina ( $\text{Al}_2\text{O}_3$ ) particles that anchor the grain boundaries for high temperature strength and creep resistance.

Critical properties of Metamic-HT have been established as minimum guaranteed values by conducting tests using ASTM sanctioned procedures (Metamic-HT Sourcebook [8.9.7]). The critical structural properties include yield strength, tensile strength, Young's modulus, and area reduction, (See Chapter 1, Section 1.2.1.4).

The neutron absorbing properties of Metamic-HT are addressed in Section 8.9 and also in Chapter 1, Section 1.2.1.4.

Chapter 3 presents structural evaluation of spent fuel basket made of Metamic-HT wherein it is concluded that the Metamic-HT plates possess adequate structural strength to meet the loadings postulated for the fuel basket. Section 8.12 presents potential for chemical and galvanic reaction in Metamic-HT under short-term and long-term operating conditions.

All Metamic-HT material procured for use in the Holtec casks is qualified as *important-to-safety* (ITS). Accordingly, material and manufacturing control processes are established to eliminate the incidence of errors, and inspection steps are implemented to serve as an independent set of barriers to ensure that all *critical characteristics* defined for the material by the cask designer are met in the manufactured product. Additional discussions on the manufacturing of Metamic-HT are provided in Chapter 1, Section 1.2.1.4 and also in Chapter 10.

### iii. Carbon Steel, Low-Alloy, and Nickel Alloy Steel

Materials for HI-STORM FW overpack and HI-TRAC VW transfer cask including the parts used to lift the overpack and the transfer cask, which may also be referred to as “significant-to-handling” or “STH” parts, are selected to preclude any concern of brittle fracture. Details of discussions are provided in Subsection 8.4.3.

Steel forging materials for low temperature applications have been selected for the STH components that have thicknesses greater than 2” so that acceptable fracture toughness at low temperatures can be assured. All other major steel structural materials in the HI-STORM FW overpack and HI-TRAC VW cask are made of fine grain low carbon steel (see drawings in Section 1.5).

The mechanical properties of these materials are provided in Section 3.3. Section 3.1 provides allowable stresses under different loading conditions and impact testing requirements for these materials.

Chapter 3 provides structural evaluations of the HI-STORM FW System components. It is demonstrated that the structural steel components of the HI-STORM FW overpack meet the allowable stress limits for normal, off-normal and accident loading conditions.

## 8.2.2 Nonstructural Materials

### i. Aluminum Alloy

The space between the fuel basket and the inside surface of the Confinement Boundary is occupied by specially shaped precision extruded or machined basket shims made of a high strength and creep resistant aluminum alloy. The basket shims establish a conformal contact interface with the fuel basket and the MPC shell, and thus prevent significant movement of the basket. The basket shims are extruded and/or machined to a precise shape with a high degree of accuracy.

The clearance between the basket shims and the interfacing machined surface of the MPC cavity is set to be sufficiently small such that the thermal expansion of the parts inside the MPC under Design Basis heat load conditions will minimize any macro-gaps at the interface and thus minimize any resistance to the outward flow of heat, while ensuring that there is no restraint of free thermal expansion.

To further enhance thermal performance, the aluminum alloy basket shims are hard anodized. This provides for added corrosion protection and to achieve the emissivity value specified in Section 4.2. Mechanical properties of the shim material are provided in Section 3.3.

The basket shim material utilized in the HI-STORM FW system has also been used in other casks (viz. HI-STAR 180).

### ii. Concrete

The plain concrete between the overpack inner and outer steel shells and in the overpack lid is specified to provide the necessary shielding properties and compressive strength. Appendix 1.D of the HI-STORM 100 FSAR which provides technical and placement requirements on plain concrete is also invoked for HI-STORM FW concrete.

The HI-STORM FW overpack concrete is enclosed in steel inner and outer shells connected to each other by radial ribs, and top and bottom plates and does not require rebar. As the HI-STORM FW overpack concrete is not reinforced, the structural analysis of the overpack only credits the compressive strength of the concrete.

The technical requirements on testing and qualification of the HI-STORM FW plain concrete are identical to those used in the HI-STORM 100 program. Accordingly, the testing and placement guidelines in Appendix 1.D of the HI-STORM 100 FSAR (Docket No. 72-1014), is incorporated in this SAR by reference.

ACI 318 is the reference code for the plain concrete in the HI-STORM FW overpack. ACI 318.1-85(05) is the applicable code utilized to determine the allowable compressive strength of the plain concrete credited in structural analysis.

The gamma shielding characteristics of concrete is considered in Section 8.8.

### iii. Lead

HI-TRAC VW contains lead between its inner and the middle shell for gamma shielding. The load carrying capacity of lead is neglected in all structural analysis. However, in the analysis of a tornado missile strike the elasto-plastic properties of lead are considered in characterizing the penetration action of the missile.

Applicable mechanical properties of lead are provided in Section 3.3. Shielding properties of lead are provided in Section 8.8.

## 8.2.3 Critical Characteristics and Equivalent Materials

As defined in the Glossary, the *critical characteristics* of a material are those attributes that have been identified, in the associated material specification, as necessary to render the material's intended function. However, material designations adopted by the International Standards Organization (ISO) also affect the type of steels and steel alloys available from suppliers around the world. Therefore, it is necessary to provide for the ability in this FSAR to substitute materials with equivalent materials in the manufacture of the equipment governed by this FSAR.

As defined in the Glossary, *equivalent materials* are those materials with critical characteristics that meet or exceed those specified for the designated material. Substitution by an equivalent material can be made after the equivalence in accordance with the provisions of this FSAR has been established.

The concept of equivalent materials explained above has been previously used in this FSAR to qualify four different austenitic stainless steel alloys (ASME SA240 Types 304, 304LN, 316, and 316LN) to serve as candidate MPC materials.

The equivalence of materials is directly tied to the notion of *critical characteristics*. A critical characteristic of a material is a material property whose value must be specified and controlled to ensure an SSC will render its intended function. The numerical value of the critical characteristic invariably enters in the safety evaluation of an SSC and therefore its range must be guaranteed. To ensure that the safety calculation is not adversely affected properties such as Yield Strength, Ultimate Strength and Elongation must be specified as *minimum* guaranteed values. However, there are certain properties where both minimum and maximum acceptable values are required (in this category lies specific gravity and thermal expansion coefficient).

Table 8.2.1 lists the array of properties typically required in safety evaluation of an SSC in dry storage and transport applications. The required value of each applicable property, guided by the safety evaluation needs defines the critical characteristics of the material. The subset of applicable properties for a material depends on the role played by the material. The role of a material in the SSC is divided into three categories:

Type	Technical Area of Applicability
S	Those needed to ensure <u>s</u> tructural compliance
T	Those needed to ensure <u>t</u> hermal compliance (temperature limits)
R	Those needed to ensure <u>r</u> adiation compliance (criticality and shielding)

The properties listed in Table 8.2.1 are the ones that may apply in a dry storage or transport application.

The following procedure shall be used to establish acceptable equivalent materials for a particular application.

Criterion i: Functional Adequacy:  
Evaluate the guaranteed critical characteristics of the equivalent material against the values required to be used in safety evaluations. The required values of each critical characteristic must be met by the minimum (or maximum) guaranteed values (MGVs of the selected material).

Criterion ii: Chemical and Environmental Compliance:  
Perform the necessary evaluations and analyses to ensure the candidate material will not excessively corrode or otherwise degrade in the operating environment.

A material from another designation regime that meets Criteria (i) and (ii) above is deemed to be an acceptable material, and hence, equivalent to the candidate material.

Equivalent materials as an alternative to the U.S. national standards materials (e.g., ASME, ASTM, or ANSI) shall not be used for the Confinement Boundary materials. Equivalent materials as alternative to Holtec's specialty engineered Metamic-HT material shall not be used for the MPC fuel basket. For other ITS materials, recourse to equivalent materials shall be made only in the extenuating circumstances where the designated material in this FSAR is not readily available.

As can be ascertained from its definition in the glossary, the *critical characteristics* of the material used in a subcomponent depend on its function. The overpack lid, for example, serves as a shielding device and as a physical barrier to protect the MPC against loadings under all service conditions, including extreme environmental phenomena. Therefore, the critical characteristics of steel used in the lid are its strength (yield and ultimate), ductility, and fracture resistance.



The appropriate critical characteristics for structural components of the HI-STORM FW System, therefore, are:

- i. Material yield strength,  $\sigma_y$
- ii. Material ultimate strength,  $\sigma_u$
- iii. Elongation,  $\epsilon$
- iv. Charpy impact strength at the lowest service temperature for the part,  $C_i$

Thus, the carbon steel specified in the drawing package can be substituted with different steel so long as each of the four above properties in the replacement material is equal to or greater than their minimum values used in the qualifying analyses used in this FSAR. The above *critical characteristics* apply to all materials used in the primary and secondary structural parts of the steel weldment in the overpack.

In the event that one or more of the *critical characteristics* of the replacement material is slightly lower than the original material, then the use of the §72.48 process shall be necessary to ensure that all regulatory predicates for the material substitution are fully satisfied.

<b>Table 8.2.1</b> <b>Critical Characteristics of Materials Required for Safety Evaluation of Storage and Transport Systems</b>				
	<b>Property</b>	<b>Type</b>	<b>Purpose</b>	<b>Bounding Acceptable Value</b>
1.	Minimum Yield Strength	S	To ensure adequate elastic strength for normal service conditions	Min.
2.	Minimum Tensile Strength	S	To ensure material integrity under accident conditions	Min.
3.	Young's Modulus	S	For input in structural analysis model	Min.
4.	Minimum elongation of $\delta_{min}$ , %	S	To ensure adequate material ductility	Min.
5.	Impact Resistance at ambient conditions	S	To ensure protection against crack propagation	Min.
6.	Maximum allowable creep rate	S	To prevent excessive deformation under steady state loading at elevated temperatures	Max.
7.	Thermal conductivity (minimum averaged value in the range of ambient to maximum service temperature, $t_{max}$ )	T	To ensure that the basket will conduct heat at the rate assumed in its thermal model	Min.
8.	Minimum Emissivity	T	To ensure that the thermal calculations are performed conservatively	Min.
9.	Specific Gravity	S (and R)	To compute weight of the component (and shielding effectiveness)	Max. (and Min.)
10.	Thermal Expansion Coefficient	T (and S)	To compute the change in basket dimension due to temperature (and thermal stresses)	Min. (and Max.)
11.	Boron-10 Content	R	To control reactivity	Min.

### 8.3 APPLICABLE CODES AND STANDARDS

The principle codes and standards applied to the HI-STORM FW System components are the ASME Boiler and Pressure Vessel Code [8.3.1], the ACI code [8.3.2], the ASTM Standards and the ANSI standards. Chapter 1 provides details of the specific applications of these codes and standards along with the other codes and standards that are applicable.

Section 1.0 of this FSAR provides a tabulation of this FSAR's compliance with NUREG-1536. This section also provides a list of clarifications and alternatives to NUREG-1536. This list of clarifications and alternatives discusses Holtec International's approach for compliance with the underlying intent of the guidance and also provides the justification for the alternative method for compliance adopted in this FSAR. Section 1.2 identifies the ASME code paragraphs applicable for the design of the HI-STORM FW overpack primary load bearing parts, summarizes the code requirements for the fabrication of the HI-STORM FW components, and refers to the national standards (e.g., ASTM, AWS, ANSI, etc.) used for the material procurement and welding.

Chapter 2 discusses factored load combinations for ISFSI pad design per NUREG-1536 [8.3.3], which is consistent with ACI-349-85. Codes ACI 360R-92, "Design of Slabs on Grade"; ACI 302.1R, "Guide for Concrete Floor and Slab Construction"; and ACI 224R-90, "Control of Cracking in Concrete Structures" are also used in the design and construction of the concrete pad. Section 2.2 elaborates on the specific applications of ASME Boiler and Pressure Vessel code and provides a list of ASME code alternatives for the HI-STORM FW System.

Section 3.1 provides allowable stresses and stress intensities for various materials extracted from applicable ASME code sections for various service conditions. This section also provides discussions on fracture toughness test requirements per ASME code sections. Mechanical properties of materials are extracted from applicable ASME sections and are tabulated for various materials used in HI-STORM FW System. Concrete properties are from ACI 318-89 code. Section 3.7 presents discussions on compliance on NUREG-1536 and stipulations of 10CFR72 requirements to provide reasonable assurance with respect to the adequacy of the HI-STORM FW System.

In order to meet the requirements of the codes and standards the materials must conform to the minimum acceptable physical strengths and chemical compositions and the fabrication procedures must satisfy the prescribed requirements of the applicable codes.

Additional codes and standards applicable to welding are discussed in Section 8.5 and those for the bolts and fasteners are discussed in Section 8.6.

Review of the above shows that the identified codes and standards are appropriate for the material control of major components. Additional material control is identified in material specifications. Material selections are appropriate for environmental conditions to be encountered during loading, unloading, transfer and storage operations. The materials and fabrication of major components are suitable based on the applicable codes of record.

## 8.4 MATERIAL PROPERTIES

This section provides discussions on material properties that mainly include mechanical and thermal properties. The material properties used in the design and analysis of the HI-STORM FW System are obtained from established industry codes such as ASME Boiler and Pressure Vessel Code [8.4.1], ASTM publications, handbooks, textbooks, other NRC-reviewed SARs, and government publications, as appropriate.

### 8.4.1 Mechanical Properties

Section 3.3 presents mechanical properties of materials used in the HI-STORM FW System. The structural materials include Alloy X, Metamic-HT, carbon steel, low-alloy and nickel-alloy steel, bolting materials and weld materials. The properties include yield stress, mean coefficient of thermal expansion, ultimate stress and the Young's modulus of these materials and their variations with temperature. Certain mechanical properties are also provided for nonstructural materials such as concrete and lead used for shielding. Additional properties of the neutron absorbing material Metamic-HT are discussed in Section 8.9.

The discussion on mechanical properties of materials in Chapter 3 provides reasonable assurance that the class and grade of the structural materials are acceptable under the applicable construction code of record. Selected parameters such as the temperature dependent values of stress allowables, modulus of elasticity, Poisson's ratio, density, thermal conductivity and thermal expansion have been appropriately defined in conjunction with other disciplines. The material properties of all code materials are guaranteed by procuring materials from Holtec approved vendors through material dedication\*, process if necessary.

### 8.4.2 Thermal Properties

Section 4.2 presents thermal properties of materials used in the MPC such as Alloy X, Metamic-HT, aluminum shims and helium gas; materials present in HI-STORM FW such as carbon steel and concrete; and materials present in HI-TRAC VW transfer cask that include carbon steel, lead and demineralized water. The properties include density, thermal conductivity, heat capacity, viscosity, and surface emissivity/absorptivity. Variations of these properties with temperature are also provided in tabular forms.

The thermal properties of fuel ( $\text{UO}_2$ ) and fuel cladding are also reported in Section 4.2.

Thermal properties are often obtained from standard handbooks and established text books (see Table 4.2.1). When variations of thermal properties are observed the most conservative values are established as input for the design of the components of the HI-STORM FW System.

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\* A term of art in nuclear quality assurance.

### 8.4.3 Low Temperature Ductility of Ferritic Steels<sup>\*</sup>

The risk of brittle fracture in the HI-STORM FW components is eliminated by utilizing materials that maintain high fracture toughness under extremely cold conditions.

The MPC canister is constructed from a menu of stainless steels termed Alloy X. These stainless steel materials do not undergo a ductile-to-brittle transition in the minimum service temperature range of the HI-STORM FW System. Therefore, brittle fracture is not a concern for the MPC components.

Such an assertion cannot be made *a priori* for the HI-STORM FW storage overpack and HI-TRAC VW transfer cask that contain ferritic steel parts. In general, the impact testing requirement for the HI-STORM FW overpack and the HI-TRAC VW transfer cask is a function of two parameters: the Lowest Service Temperature (LST)<sup>†</sup> and the normal stress level. The significance of these two parameters, as they relate to impact testing of the overpack and the transfer cask, is discussed below.

In normal storage mode, the LST of the HI-STORM FW storage overpack structural members may reach -40°F in the limiting condition wherein the spent nuclear fuel (SNF) in the contained MPCs emits no (or negligible) heat and the ambient temperature is at -40°F (design minimum per Chapter 2: Principal Design Criteria). However, during the HI-STORM FW overpack transport operations, the applicable lowest service temperature is per 0°F (per the Technical Specifications). Therefore, two distinct LSTs are applicable to load bearing metal parts within the HI-STORM FW System; namely,

LST = 0°F for the HI-STORM FW overpack during transport operations and for the HI-TRAC VW transfer cask during all normal operating conditions.

LST = -40°F for the HI-STORM FW overpack during storage operations.

SA350-LF2 and SA350-LF3 have been selected as the material for the STH parts due to their capability to maintain acceptable fracture toughness at low temperatures (see Table 5 in SA350 of ASME Section IIA).

Table 3.1.9 provides a summary of impact testing requirements for the materials used in the HI-STORM FW System to ensure prevention of brittle fracture.

### 8.4.4 Creep Properties of Materials

Creep, a visco-elastic and visco-plastic effect in metals, manifests itself as a monotonically increasing deformation if the metal part is subjected to stress under elevated temperature. Since

<sup>\*</sup> This subsection has been copied from the HI-STORM 100 FSAR (Section 3.1) without any substantive change.

<sup>†</sup> LST (Lowest Service Temperature) is defined as the daily average for the host ISFSI site when the outdoors portions of the “short-term operations” are carried out.

certain parts of the HI-STORM FW System, notably the fuel basket, operate at relatively high temperatures, creep resistance of the fuel basket is an important property. Creep is not a concern in the MPC enclosure vessel, the HI-STORM FW overpack, or the HI-TRAC VW steel weldment because of the operating metal temperatures, stress levels and material properties. Steels used in ASME Code pressure vessels have a high threshold temperature at which creep becomes a factor in the equipment design. The ASME Code Section II material properties provide the acceptable upper temperature limit for metals and alloys acceptable for pressure vessel service. In the selection of steels for the HI-STORM FW System, a critical criterion is to ensure that the sustained metal temperature of the part made of the particular steel type shall be less than the Code allowable temperature for pressure vessel service (ASME Section III Subsection ND). This criterion guarantees that excessive creep deformation will not occur in the steels used in the HI-STORM FW System.

As discussed below, the incidence of creep in the fuel basket is a not a trivial matter because lateral creep deformation can alter the reactivity control characteristics of the basket.

#### 8.4.4.1 Metamic-HT

Metamic-HT is the sole constituent material in the HI-STORM FW fuel basket. The suitability of Metamic-HT for the conditions listed in Table 8.1.1 are considered in the “Metamic-HT Qualification Sourcebook” [8.9.7].

The Metamic Sourcebook contains data on the testing to determine the creep characteristics of the Metamic-HT under both unirradiated and irradiated conditions. A creep equation to estimate a bounding estimate of total creep as a function of stress and temperature is also provided. The creep equation developed from this test provides a conservative prediction of accumulated creep strain by direct comparison to measured creep in unirradiated and irradiated coupons.

The creep equation for Metamic-HT that bounds *all* measured data (tests run for 20,000 hours) is of the classical exponential form in stress and temperature (see Table 1.2.8) stated symbolically  $\epsilon = f(\sigma, T)$ .

Creep in the fuel basket will not affect reactivity because the basket is oriented vertically during all operations (with the rare handling exception of the transfer cask as described in Subsection 4.5.1). The lateral loading of the fuel basket walls is insignificant and hence there is no mechanistic means for the basket panels to undergo lateral deformation from creep.

The creep effect would tend to shorten the fuel basket under the self-weight of the basket. An illustrative calculation of the cumulative reduction of the basket length is presented below to demonstrate the insignificant role of creep in the fuel basket.

The in-plane compressive stress,  $\sigma$ , at height  $x$  in the basket panel is given by

$$\sigma = \rho(H-x) \quad (8.1)$$

Where

$\rho$  = density of Metamic-HT

$H$  = height of the fuel basket

Using the above stress equation, the total creep shrinkage,  $\delta$ , is given by

$$\delta = \int_0^{\tau^*} \left\{ \int_0^H (\sigma, T) dx \right\} d\tau \quad (8.2)$$

Where

$T$  = panel's metal temperature, initial value conservatively assumed to be 350°C (from Section 4.6) and dropping linearly to 150°C at 60 years.

$\tau^*$  = 60 years

$H$  = height of the basket (approximately 200 inches)

Using the creep equation [1.2.6] and performing the above double integration numerically with Mathcad yields  $\delta = 0.044$  inch. In other words, the computed shrinkage of the basket is less than 0.022% of its original length.

It is concluded that for vertical configuration of storage the creep effects of the MPC basket are insignificant due to absence of any meaningful loads on the panels. Therefore, creep in the Metamic-HT fuel basket is not a matter of safety concern.

#### 8.4.4.2 Aluminum Alloy

The basket shims are not subject to any significant loading during storage. Similar to the fuel basket, the stress levels from self-weight in long-term storage eliminates creep as a viable concern for the basket shims.

## 8.5 WELDING MATERIAL AND WELDING SPECIFICATION

Welds in the HI-STORM FW System are divided into two broad categories:

- i. Structural welds
- ii. Non-structural welds

Structural welds are those that are essential to withstand mechanical and inertial loads exerted on the component under normal storage and handling.

Non-structural welds are those that are subject to minor stress levels and are not critical to the safety function of the part. Non-structural welds are typically located in the redundant parts of the structure. The guidance in the ASME Code Section NF-1215 for secondary members may be used to determine whether the stress level in a weld qualifies it to be categorized as non-structural.

Both structural and non-structural welds must satisfy the material considerations listed in Tables 8.1.1, 8.1.2, and 8.1.3, for the MPC, the HI-STORM FW overpack and the HI-TRAC VW transfer cask, respectively. In addition, the welds must not be susceptible to any of the applicable failure modes in Table 8.1.4.

To ensure that all welds in the HI-STORM FW System shall render their intended function, the following requirements are observed:

- i. The weld joint configuration is selected to accord with the function of the joint (Holtec Position Paper DS-329 [8.5.1] provided to the USNRC in Docket No. 72-1014).
- ii. The welding procedure specifications comply with ASME Section IX for every Code material used in the system.
- iii. The quality assurance requirements applied to the welding process correspond to the highest ITS classification of the parts being joined.
- iv. The non-destructive examination of every code weld is carried out using quality procedures that comply with ASME Section V.
- v. Metamic-HT welding and welder qualifications, requirements, and examinations will be in accordance with Paragraphs 10.1.6.2, 10.1.1.4, and the drawing package in Section 1.5.

The welding operations are performed in accordance with the requirements of codes and standards depending on the design and functional requirements of the components.



The selection of the weld wire, welding process, range of essential and non-essential variables,\* and the configuration of the weld geometry has been carried out to ensure that each weld will have:

- i. Greater mechanical strength than the parent metal.
- ii. Acceptable ductility, toughness, and fracture resistance.
- iii. Corrosion resistance properties comparable to the parent metal.
- iv. No risk of crack propagation under the applicable stress levels.

The welding procedures implemented in the manufacturing of HI-STORM FW System components are intended to fulfill the above performance expectations.

Additional information on the welding for HI-STORM FW System components is provided in Section 1.2. Lists of codes and standards applicable for the manufacturing of HI-STORM FW System are also provided therein.

A list of ASME code alternatives for the MPC fabrication including welding is presented in Section 2.2. The structural strength requirements of welds including fracture toughness test requirements of weld materials are provided in Section 3.1. The confinement boundary welds and their testing requirements are discussed in Section 7.1. The inspection and testing requirements of the HI-STORM FW System component welds are provided in Section 10.1.

The weld filler material shall comply with requirements set forth in the applicable Welding Procedure Specifications qualified to ASME Section IX at the manufacturer's facility. Only those welding procedures that have been qualified to the Code are permitted in the manufacturing of HI-STORM FW components.

Review of the above shows that except for the MPC lid welds, all welds of the Enclosure Vessel are full penetration weld with volumetric NDE. All weld filler metals are specified by ASME Section II, Part C and associated AWS classification in applicable weld procedures.

The weld procedure qualification record specifies the requirements for fracture control (e.g. post weld heat treatment). The HI-STORM FW overpack and HI-TRAC VW transfer cask do not require any post weld heat treatment due to the material combinations and provisions in the applicable codes and standards. With respect to the MPC Lid-to-Shell weld, the progressive P.T. requirements on the shell/lid weld are identical to those in Docket No. 72-1014 (which are derived from the analysis summarized in Holtec Position Paper DS-213 [8.5.2], provided to the USNRC on Docket No. 72-1014.

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\* Please refer to Section IX of the ASME Code for the definition and delineation of essential and non-essential variables.

Non-structural welds shall meet the following requirements:

1. The welding procedure shall comply with Section IX of the ASME Code or AWS D1.1.
2. The welder shall be qualified, at minimum, to the commercial code such as ASME Section VIII, Div.1, or AWS D1.1.
3. The weld shall be visually examined by the weld operator or a Q.C. inspector qualified to Level 1 (or above) per ASNT designation.

## 8.6 BOLTS AND FASTENERS

Chapter 3 provides information on the structural evaluation of the bolts and fasteners. Section 3.1 discusses fracture toughness requirements for bolting materials. Section 3.3 provides the bolting materials used in the HI-STORM FW System. Section 3.3 (Table 3.3.4) provides mechanical properties of bolting materials.

Chapter 9 provides pre-tensioning requirements for HI-STORM FW System bolts to ensure that the bolts shall not be overstressed under any condition of loading applicable to the system.

Bolts and fasteners made of low alloy steel are not expected to experience any significant corrosion in the operating environment. The ISFSI operation and maintenance program shall call for coating of bolts and fasteners if the ambient environment is aggressive.

A review of the above shows that the materials for the bolts and the fasteners have been selected to possess the required tensile strengths, resistance to corrosion and brittle fracture. To prevent a change in the bolt pre-stress during operating conditions, the coefficient of thermal expansion of each bolt material has been closely matched to that of the parts being fastened together.

Preventing galling of interfacing surfaces is another critical consideration in selecting bolt materials. Use of austenitic stainless bolts on interfacing austenitic stainless steel surfaces is not permitted. All threaded surfaces are treated with a preservative to prevent corrosion. The O&M program for the storage system calls for all bolts to be monitored for corrosion damage and replaced, as necessary.

## 8.7 COATINGS

Protective coatings are used primarily as a corrosion barrier and/or as a means to facilitate decontamination. Coating materials for the HI-STORM FW system components are guided by the successful experience in similar service applications of the HI-STORM 100 and HI-STAR 100 components and parts. The main considerations in the selection of coatings are the ruggedness and physical integrity in the specific service environment, ease of decontamination as applicable to immersion service, thermal and radiation stability, and ease of application to facilitate touch-up activities for preventive maintenance. Surface preparation and repair are performed in accordance with manufacturer recommendations.

The coatings applied on specific HI-STORM FW System components are selected to be compatible with their respective conditions of service. For example, equipment used in the fuel pool environment must be conducive to convenient decontamination. Protective coatings are applied to surfaces vulnerable to corrosion such as exposed carbon steel surfaces on the HI-STORM FW overpack and HI-TRAC VW transfer cask. The MPC surfaces are not coated.

### 8.7.1 Environmental Conditions Applicable to Coating Selection and Evaluation Criteria:

#### 8.7.1.1 Environmental Conditions

The environmental conditions that warrant consideration in the selection of coatings are:

- i. Temperature, humidity, and insolation
- ii. Radiation field
- iii. Immersion service

Temperature, humidity, and insolation conditions may vary at different ISFSI sites. The coating selected for the HI-STORM FW overpack, which is subject to long-term exposure, must be stable under the entire range of psychrometric conditions that prevail in the territorial United States. The coating selected for HI-TRAC VW must withstand the thermal exposure during fuel drying operations and during immersion in the spent fuel pool.

Stable performance under radiation is important for coatings applied on the inside surfaces of the HI-STORM FW overpack and the HI-TRAC VW transfer cask, which are proximate to the lateral surfaces of the MPC.

Immersion in the pool implies three major challenges to the coating on the HI-TRAC VW:

- a. Risk of penetration of tiny contaminant particulates in the pores of the coating.
- b. Chemical attack (by boric acid in PWR pools and demineralized water in BWR pools).
- c. Temperature change as the transfer cask is immersed in or withdrawn from water.

Coatings that have been determined to be unsuitable for the immersion service shall not be used in the HI-TRAC VW transfer cask.

#### 8.7.1.2 Coating Evaluation Criteria

The evaluation criteria for selecting coatings are summarized below. These criteria shall be used if a pre-approved coating listed in Subsection 8.7.2, for any reason, is no longer available for use.

<b>Coating Acceptance Criteria</b>	
1.	Non-reactive to the surrounding environment
2.	Structural performance (bendability, ductility, resistance to cracking, and resistance to abrasion)
3.	Adherence to base material
4.	Chemical immersion resistance, if applicable
5.	Emissivity and absorptivity consistent with thermal analysis
6.	Temperature resistance for analyzed temperature conditions with humidity and insolation, as applicable
7.	Radiation resistance for analyzed conditions

The paint suppliers may certify the properties by performance of applicable ASTM tests. In the absence of ASTM test data for a required characteristic in the above table, the coating supplier will provide evidentiary information to justify acceptance. Alternatively, Holtec International will perform its own independent tests to establish compliance with the required criteria.

#### 8.7.2 Acceptable Coatings

Proven (previously used on HI-STORM 100 System components and other cask designs) coatings and paints that adequately satisfy the requirements are presented below and pre-approved for use on HI-STORM FW System components.

Carboguard 890 (Cycloaliphatic Amine Epoxy) of Carboline Company which demonstrates acceptable performance for short-term exposure in mild borated pool water may be used for coating the HI-TRAC VW transfer cask exterior surfaces as well as HI-STORM FW overpack surfaces. This coating is certified for immersion services and provides excellent chemical resistance and abrasion resistance. It provides a smooth surface with no porosity and thereby, excellent decontamination characteristics. No adverse interaction has been experienced in many years of use.

Thermaline 450 (Amine-Cured Novolac Epoxy) of Carboline Company may be used for coating HI-TRAC VW transfer cask internal surfaces which are exposed only to demineralized water during in-pool operations (the annulus is filled prior to placement in the spent fuel pool and the inflatable seal prevents fuel pool water in-leakage) and higher service temperatures. This coating provides excellent resistance to corrosion, abrasion, and permeation. No adverse interaction has been experienced in many years of use.

Carbozinc 11 (also known as CZ-11) may be used for coating HI-STORM FW overpack internal cavity and external surfaces (including lid surfaces). This solvent based coating material has excellent corrosion resistant properties in harsh environments and provides inorganic zinc (galvanic) protection to steel surfaces. As an alternative to the Carbozinc 11, Sherwin Williams Zinc Clad II HS, Sherwin Williams Zinc Clad II Plus may also be used.

Product information for the above coatings is provided in Appendix 8.A.

Coatings that are specified in this section shall not be substituted with another coating unless the substitute meets or exceeds the performance of the coating listed above under all the applicable coating evaluation criteria set forth in the previous subsection.

### 8.7.3 Coating Application

Holtec utilizes Q.A.-validated written procedures (HSP-318 [8.7.1] and HSP-319 [8.7.2]) to achieve the desired performance for the coating. These procedures provide requirements for the preparation and painting of the HI-STORM FW overpack, HI-TRAC VW transfer cask and associated components. These procedures are based on paint manufacturers' applicable specifications, instructions and recommendations.

The procedures provide details for the preparation prior to blasting, surface preparation, mixing and application, painting in the field, and touch up steps or repairs. The procedures also provide details of the dry film thickness testing and the acceptance criteria. Painting documentation is maintained for the record of the completion of various painting steps and the environmental conditions including the ambient temperature, humidity and the component surface temperature.

## 8.8 GAMMA AND NEUTRON SHIELDING MATERIALS

Gamma and neutron shield materials in the HI-STORM FW System are discussed in Section 1.2. The primary shielding materials used in the HI-STORM FW system, like the HI-STORM 100 system, are plain concrete, steel, lead, and water.

The plain concrete enclosed by cylindrical steel shells, a thick steel baseplate, and a top annular plate provides the main shielding function in the HI-STORM FW overpack. The overpack lid has appropriate concrete shielding to provide neutron and gamma attenuation to minimize skyshine.

The transfer cask in the HI-STORM FW system (HI-TRAC VW) is provided with steel and lead shielding to ensure that the radiation and exposure objectives of 10CFR72.104 and 10CFR72.106 are met. The space between the inner shell and the middle shell is occupied by lead, conforming to ASTM B29, which provides the bulk of the cask's (gamma) radiation shielding capability. The water jacket between the middle shell and the outermost shell (filled with demineralized water or ethylene glycol fortified water, depending on the site environmental constraints) provides most of the neutron shielding capability to the cask. The water in the water jacket serves as the neutron shield on demand: When the cask is in the pool and the MPC is full of water, the water jacket is kept empty (or partially empty as necessary) to minimize the cask's weight, the neutron shielding function being provided by the water in the MPC cavity. However, when the MPC is emptied of water at the Decontamination and Assembly Station (DAS), then the neutron shielding capacity of the cask is replenished by filling the water jacket. The HI-TRAC VW bottom lid is extra thick steel to provide an additional measure of gamma shielding to supplement the gamma shielding at the bottom of the MPC.

### 8.8.1 Concrete

Appendix 1.D of HI-STORM 100 FSAR provides details of the concrete properties and the testing requirements. The *critical characteristics* of concrete are its density and compressive strength.

The density of plain concrete within the HI-STORM FW overpack is subject to a minor decrease due to long-term exposure to elevated temperatures. The reduction in density occurs primarily due to liberation of unbonded water by evaporation.

The density of concrete has been classified into three states in the published literature [8.8.1].

- a) fresh density: the density of freshly mixed concrete
- b) air-dry density: drying in air under ambient conditions, where moisture is lost until a quasi-equilibrium is reached
- c) oven-dry density: concrete dried in an oven at 105°C (221°F)

Because the bulk temperature of concrete in HI-STORM FW is spatially variable, the oven-dry density is conservatively used as the reference density for shielding analysis.

Density loss during the initial drying process is considered in the fabrication of the HI-STORM FW overpack by providing wet concrete densities above the minimum required dry (hardened paste) density. Density loss during drying is on the order of 1% and conservatively imposes a larger delta between wet density and the minimum dry density. The data in the literature, viz., Neville [8.8.1] indicates that the density difference between the air-dry condition and oven-dry condition is about one fourth of the density difference experienced during the drying process. Therefore, the loss in density would be expected to be on the order of 0.25%. This density loss is very low and is considered too small to have a significant impact on the shielding performance of the overpack. Thus, the minimum “fresh density” during concrete placement is set equal to the reference density (Table 1.2.5) plus 1.25%.

Section 5.3 considers the minimum density requirements of concrete for effective shielding. The density requirement is confirmed per Appendix 1.D of the HI-STORM 100 FSAR.

## 8.8.2 Steel

Section 5.3 provides a discussion on steel as a shielding material and its composition used in the evaluation of its shielding characteristics.

## 8.8.3 Lead

Section 1.2 provides a discussion on lead used in HI-TRAC VW for gamma shielding. In the HI-TRAC VW transfer cask radial direction, gamma and neutron shielding consists of steel-lead-steel and water, respectively. In the HI-TRAC VW bottom lid, layers of steel-lead-steel provide an additional measure of gamma shielding to supplement the gamma shielding at the bottom of the MPC.

Mechanical properties of lead are provided in Section 3.3. Section 5.3 provides the minimum density and composition (mass fraction of trace elements) of lead.

## 8.8.4 Water

Water is used as a neutron shield in the HI-TRAC VW transfer cask. Section 5.3 provides the minimum density requirements of water for transfer cask water jacket and inside MPC. The shielding effectiveness is calculated based on the minimum water density at the highest operating temperature. Calculations show that additives for freeze protection (at low temperature operation) such as ethylene glycol do not have any adverse effect on effectiveness of the neutron shielding function of water in the water jacket.

As discussed in Section 5.1, there is only one accident that has any significant impact on the shielding configuration. This accident is the postulated loss of the neutron shield (water) in the HI-TRAC VW. The change in the neutron shield was conservatively analyzed by assuming that the entire volume of the liquid neutron shield was replaced by air.



## 8.9 NEUTRON ABSORBING MATERIALS

Inside the MPC enclosure vessel is a structure referred to as the fuel basket. The fuel basket is an egg-crate assemblage of Metamic-HT plates which creates prismatic cells with square cross sectional openings for fuel storage. Metamic-HT is the neutron absorber and structural material of the MPC fuel basket. Metamic-HT is a composite material of nano-particles of aluminum oxide (alumina) and finely ground boron carbide particles dispersed in a metal matrix of pure aluminum [8.9.7].

### 8.9.1 Qualification and Properties of Metamic-HT

The qualification and properties of Metamic-HT are presented in Chapter 1, Section 1.2.1.4 where its key characteristics necessary for insuring nuclear reactivity control, thermal, and structural performance are discussed. A test program configured to address the Metamic-HT properties was conducted by Holtec International and the minimum guaranteed values (MGVs) of the *critical characteristics* of Metamic-HT were determined [8.9.7] and summarized in Chapter 1, Section 1.2.1.4. All testing was conducted in accordance with the applicable ASTM test standards. The role in the fuel basket safety function of each of the critical characteristics is provided in Chapter 1, Section 1.2.1.4.

A rigorous quality control regimen and Holtec QA procedures ensure that all extruded Metamic-HT plates meet the requirements for the quality genre of the casks.

To ensure that the manufactured Metamic material will render its intended function with reasonable assurance, a sampling plan based on Mil Standard 105E [8.9.8] has been specified and made a part of the Metamic-HT Manufacturing Manual [8.9.6]. The Sampling plan shall provide a reasonable level of confidence that the Minimum Guaranteed Values of all critical mechanical properties will be met in the production lots. Additional information regarding manufacturing of Metamic-HT is provided in Chapter 1, Section 1.2.1.4.

Chapter 2 provides discussions on criticality parameters for design basis SNF, and the controls and methods utilized for prevention of criticality.

Criticality evaluation is presented in Chapter 6. The material heterogeneity parameters are adequately characterized and controlled and the criticality calculations employ appropriate corrections when modeling the heterogeneous material as an idealized homogeneous mixture. It is demonstrated that the MPC provides 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  $k_{\text{eff}} < 0.95$  for fresh unirradiated fuel with optimum water moderation and close reflection, including all biases, uncertainties, and MPC manufacturing tolerances. Additional neutronic properties of Metamic-HT are provided in Chapter 1, Section 1.2.1.4.

## 8.9.2 Consideration of Boron Depletion

The effectiveness of the 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 design life of the MPC. Analysis discussed in Section 6.3 demonstrates that the boron depletion in the neutron absorber material is negligible over the expected service life of the HI-STORM FW System. This is due to the fact that the borated material is subjected to a relatively low neutron flux. Analyses show that the depletion of boron is a small fraction of the quantity present. Therefore, sufficient levels of boron will remain in the fuel basket neutron absorbing material to maintain criticality safety functions over the design life of the MPC. Furthermore, the boron content of Metamic-HT 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 10% (see Chapter 6) for conservatism in the analysis.

## 8.10 CONCRETE AND REINFORCING STEEL

The HI-STORM FW System does not utilize concrete with rebar. The plain concrete used in the HI-STORM FW overpack serves as the neutron shielding. The absence of rebar in the HI-STORM FW overpack concrete ensures that radiation streaming paths due to the development of cracks and discontinuities at the rebar/concrete interfaces will not develop. Concrete in the overpack is not considered as a structural member, except to withstand compressive, bearing, and penetrant loads. Therefore the mechanical behavior of concrete must be quantified to determine the stresses in the structural members (steel shells surrounding it) under accident conditions.

Section 3.3 provides the concrete mechanical properties. Allowable, bearing strength in concrete for normal loading conditions is calculated in accordance with ACI 318-05 [8.3.2]. The procedure specified in ASTM C-39 is utilized to verify that the assumed compressive strength will be realized in the actual in-situ pours. Appendix 1.D in the HI-STORM 100 FSAR provides additional information on the requirements on plain concrete for use in HI-STORM FW storage overpack.

To enhance the shielding performance of the HI-STORM FW storage overpack, high density concrete can be used during fabrication. The permissible range of concrete densities is specified in Table 1.2.5.

Review of the above shows that the HI-STORM FW System concrete components are acceptable. All concrete is either encased in steel or covered underneath the overpack lid, therefore; it is not subject to weathering or other atmospheric degradation, even in marine environments. To ensure that the concrete performs its primary function (shielding integrity/effectiveness) tests are performed as required by Chapter 10.

## 8.11 SEALS

The HI-STORM FW System does not rely upon mechanical seals for maintaining the integrity of the Confinement Boundary. The MPC Vent/Drain caps washers are made of a soft and malleable metal such as aluminum 1100.

The HI-TRAC VW transfer cask bottom lid utilizes a gasket to prevent ingress of pool water when the cask is staged in the fuel pool and leakage during MPC processing operations. Gaskets used may be silicone, neoprene, and a similar elastomeric material that is inert in the pool's aqueous environment.

In selecting the gasket material, it is necessary to ensure that none of the following materials will leach out in the pool water in measurable quantities.

- Viton
- Saran
- Silastic L8-53
- Teflon
- Nylon
- Carbon steel
- Neoprene or similar materials made of halogen containing elastomers
- Rubber bonded asbestos
- Polyethelene film colored with pigments over 50 ppm fluorine, measurable amount of mercury or halogens, or more than 0.05% lead
- Materials containing lead, mercury, sulfur, phosphorus, zinc, copper and copper alloys, cadmium, tin, antimony, bismuth, mischmetal, magnesium oxide, and halogens exceeding 75 ppm (including cleaning compound).

The gaskets used in the HI-TRAC VW shall be the same or equivalent to those that have proven to be satisfactory in prior service (such as in other Holtec transfer casks).

The mechanical design details of the gasketed joint in the transfer cask follow the guidelines in Chapter 3 of [8.11.1], which recommend joints subjected to cyclic loadings to be made of the "controlled compression" genre. The "controlled compression" joint minimizes cyclic damage to the gasket.

The O&M program for the storage system calls for HI-TRAC VW transfer cask elastomeric seals to be inspected for damage and replaced on an appropriate schedule as recommended by the manufacturer.

## 8.12 CHEMICAL AND GALVANIC REACTIONS

The materials used in the HI-STORM FW System are examined to establish that these materials do not participate in any chemical or galvanic reactions when exposed to the various environments during all normal operating conditions and off-normal and accident events.

The following acceptance criteria for chemical and galvanic reactions are extracted from ISG-15 [8.1.1] for use in HI-STORM FW components.

- a. The DCSS should prevent the spread of radioactive material and maintain safety control functions using, as appropriate, noncombustible and heat resistant materials.
- b. A review of the DCSS, its components, and operating environments (wet or dry) should confirm that no operation (e.g., short term loading/unloading or long-term storage) will produce adverse chemical and/or galvanic reactions, which could impact the safe use of the storage cask.
- c. Components of the DCSS should not react with one another, or with the cover gas or spent fuel, in a manner that may adversely affect safety. Additionally, corrosion of components inside the containment vessel should be effectively prevented.
- d. The operating procedures should ensure that no ignition of hydrogen gas should occur during cask loading or unloading.
- e. Potential problems from general corrosion, pitting, stress corrosion cracking, or other types of corrosion, should be evaluated for the environmental conditions and dynamic loading effects that are specific to the component.

The materials and their ITS pedigree are listed in the drawing package provided in Section 1.5. The compatibility of the selected materials with the operating environment and to each other for potential galvanic reactions is discussed in this section.

### 8.12.1 Operating Environments

During fuel loading, handling or storage the components of the HI-STORM FW System experience the following environments (see Tables 8.1.1, 8.1.2, and 8.1.3).

- Spent Fuel Pool Water – During the fuel loading steps, the MPC confinement space is flooded with water (borated water in PWRs and demineralized water in BWRs). As water is withdrawn from the MPC space, the temperature of its contents rises, facilitating an Arrhenius-like acceleration of any chemical reaction that may occur in the presence of water and water vapor or boric acid (in PWRs). These same conditions would exist in the event an MPC needs to be unloaded and the MPC is reflooded prior to lid removal.

- Helium – During loading operations, all water is removed from the interior of the MPC and an inert gas is injected. Internal MPC components get exposed to dry helium under pressure during storage.
- External atmosphere – During long term storage the casks are exposed to outside atmosphere, air with temperature variations, solar radiation, rain, snow, ice, etc.

As discussed below, the components of the HI-STORM FW System has been engineered to ensure that the environmental conditions expected to exist at nuclear power plant installations do not prevent the cask components from rendering their respective intended functions.

## 8.12.2 Compatibility of MPC Materials

### 8.12.2.1 MPC Confinement Boundary Materials

#### Austenitic Stainless Steels

The MPC confinement boundary is composed entirely of corrosion-resistant austenitic stainless steel. 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 available austenitic stainless steels are AISI Types 304, 304LN, 316 and 316LN containing a minimum of 16% chromium and 8% nickel, and at least traces of molybdenum. The passive films (formed due to atmospheric exposure) of stainless steels range between 10 to 50 angstroms ( $1 \times 10^{-6}$  to  $5 \times 10^{-6}$  mm) thick [8.12.4]. Of all types of stainless steels (i.e., austenitic, ferritic, martensitic, precipitation hardenable and two-phase), “the austenitic stainless alloys are considered the most resistant to industrial atmospheres and acid media” [8.12.4].

The MPC contains no gasketed, threaded, or packed joints for maintaining confinement. The all-welded construction of the MPC confinement boundary and the inert backfill gas within ensures that the interior surfaces and the MPC internals (Metamic-HT baskets, shims, etc.) are not subject to corrosion. Exterior MPC surfaces would be exposed to the ambient environment while inside of a HI-STORM FW storage overpack or a HI-TRAC VW transfer cask.

#### Austenitic Stainless Steels in Demineralized and Borated Water Environments

The average MPC may be in contact with borated and/or demineralized water at temperatures below boiling and at pressures of up to three atmospheres (not including hydrotest) for approximately 2 to 3 days. For PWRs, the soluble boron levels are typically maintained at or below 2,500 ppm (0.25% boric acid solution). Experimental corrosion data for AISI Type 304 and 316 stainless steels (Swedish Designations SIS-14-2333 and SIS-14-2343, respectively) are available from the Swedish Avesta Jernverk laboratory [8.12.4]. Corrosive media evaluated in these tests include 4% (40,000 ppm) and 20% (200,000 ppm) boric acid solutions and water, all at boiling. Under the evaluated conditions, the tested steels are identified as “fully resistant”, with corrosion rates of less than 0.1 mm per year. Even more extensive experimental corrosion

data is available from ASM International [8.12.1]. For test conditions without rapid agitation, similar to conditions that would exist during MPC fuel loading in a spent fuel pool, all austenitic stainless steels available for MPC fabrication (i.e., AISI Types 304, 304LN, 316 and 316LN) are extremely resistant to corrosion in boric acid and water. More specifically, one set of data (UNS No. S30400) for 2.5% boric acid solution and water at 90.6°C (195°F), under no aeration and rapid agitation yielded a maximum corrosion rate of 0.003 mm per year [8.12.1].

No structural effects from any potential corrosion from demineralized and borated water environments are expected. Loading of a dry storage cask with reasonable delays can take up to two weeks. Adjusting the worst-case data for a 0.25% boric acid concentration the maximum thinning of any structural member in an MPC is only  $4.80 \times 10^{-6}$  mm (1.89 microinches). This is a negligibly small fraction (0.0006%) of the thickness of the thinnest structural member 7.9 mm (0.3125 in.) and a negligibly small fraction (0.004%) of the tolerance on the material thickness (0.045 in.) permitted by the governing ASME Code [8.12.2].

#### Austenitic Stainless Steels and Crud

Corrosion products cause “crud” deposits on fuel assemblies. Industry experience shows that crud, which is stable in oxygenated solutions, has not been found to contain materials that can react with stainless steel and cause significant degradation. Crud may leave a slight film of rust on the interior surfaces of the MPC during fuel loading and closure activities.

#### Austenitic Stainless Steels and Boron Crystals

Dry boron or boric acid crystals that remain in the MPC after drying and helium backfill are expected to have negligible corrosive effects on stainless steel due to the absence of the necessary reagents (oxygen and moisture).

#### Austenitic Stainless Steels and Marine Environments

The MPC is designed to be loaded with spent fuel assemblies from most light water reactor (LWR) nuclear power plants. LWR nuclear power plants, in general, are located near large bodies of water to ensure an adequate supply of cooling water. As a result many nuclear power plants and, subsequently, many potential ISFSI sites are located in coastal areas where dissolved salts may be present in atmospheric moisture. Casks deployed at coastal ISFSI sites that would be exposed to the harsh marine environment for prolonged periods must not suffer corrosion that will impair their functionality.

Extensive data show corrosion rates (pitting) to 0.0018 (mm/yr) for 304, 304LN, 316 and 316LN in marine environments at ambient temperatures after 26 years [8.12.1]. Using this bounding corrosion rate data, a Holtec Position Paper [8.12.3] estimates the total corrosion of the external surface of the MPC in 100 years of service is about half a millimeter which is significantly smaller than the available design margins in the material thickness. It is to be noted that this

upper-bound is estimated for an extreme hypothetical marine environment. As discussed earlier for inland applications the corrosion rates are insignificant.

Therefore, corrosion of the MPC in long-term storage is not a credible safety concern.

#### Austenitic Stainless Steels and Hydrogen Damage

Traces of hydrogen may be present under the MPC Lid during welding operations. The hydrogen content is limited due to a low hydrogen generation rate and the (required) purging of the underside of the lid with helium. Hydrogen damage is classified into four distinct types (1) hydrogen blistering, (2) hydrogen embrittlement, (3) decarburization, (4) hydrogen attack. Decarburization and hydrogen attack are high temperature processes and therefore may be of concern during cooling of the weld puddle. Austenitic stainless steels are one of the few metals that perform satisfactorily at all temperatures and pressures in the presence of hydrogen [8.12.6]. Considering the limited hydrogen concentration, limited time (2-3 days) for fuel loading and limited pressures and temperatures (with the exception of high temperatures at the lid to shell weld), hydrogen damage is not an applicable corrosion mechanism during fuel loading. With respect to the lid to shell weld, the weld design, use of a continuous inert gas purge, the weld method and NDE inspections provide assurance that the weld has no credible damage and is of high integrity.

#### 8.12.2.2 Materials of MPC Internals

The internals of the MPC consists of Metamic-HT fuel baskets and aluminum alloy shims for basket support. Besides these internals, SNF, possible failed fuel and/or damaged fuel with containers, and non-fuel hardware, a sealed MPC may also contain boric acid crystals (in PWRs) and crud. The cleanliness requirements and inspections during fabrication and fuel loading operations ensure that the MPC has minimal surface debris and impurities.

#### Tests on Metamic-HT

Extensive tests [8.9.7] have been conducted to establish material properties of Metamic-HT including its corrosion-resistance characteristics. The Metamic-HT specimens were used for corrosion testing in demineralized water and in 2000 ppm boric acid solution. The tests concluded that the Metamic-HT panels will sustain no discernible degradation due to corrosion when subjected to the severe thermal and aqueous environment that exists around a fuel basket during fuel loading or unloading conditions.

#### Aluminum Alloy

Aluminum alloy used in the fuel basket shims are hard anodized. The anodizing is an electrolytic passivation process used to increase the thickness of the natural oxide layer on the surface of metal parts. Anodizing increases corrosion resistance and wear resistance of the material surface. There is no mechanistic process for the basket shims with hard anodized surface to react with



borated water or demineralized water during fuel loading operation. Under the long-term storage condition, the basket shims are exposed to dry and inert helium with no potential for reaction.

#### Effect of Forced Helium Dehydration (FHD) Process

The operation of the FHD consists of flowing hot dry helium through the MPC at pressures and temperature limited by the MPC design pressure and temperature of the MPC. Due to the purity of the helium stream and the relatively short duration (normally 10 to 60 hours), no significant corrosion mechanisms are identified.

#### 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 (see 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 and the associated fabrication and closure requirements for the canister are specifically set down to assure that an inert helium atmosphere is maintained in the canister throughout the MPC's 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 the 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.

#### 8.12.2.3 Galvanic Corrosion

The MPC is principally constructed of stainless steel shell and Metamic-HT. Borated aluminum 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 Metamic (classic) and stainless steel materials. Not one case of chemical or galvanic degradation has been found in such fuel racks. This experience provides a sound basis to conclude that corrosion will not occur in these materials. For further protection, both Metamic-HT and aluminum basket shims are installed in the anodized state in the MPC.

Furthermore, galvanic corrosion is not an applicable mechanism since the interior of the MPC during normal operation is essentially devoid of any moisture and the MPC shell surfaces are expected to be practically free from condensation. Finally, the interior of the carbon steel HI-STORM FW overpack is painted to inhibit corrosion.

During long-term storage in the HI-STORM FW overpack, the MPC operates at elevated temperatures under normal conditions while inside the HI-STORM. The external ambient environment normally consists of atmospheric conditions, which include humidity and perhaps airborne contaminants such as sulfur dioxide, chlorine gas, sulfur gas and ozone. The interior is backfilled with highly pure helium. The spent fuel irradiates the MPC but at much lower levels than those experienced in an operating reactor. It is recognized that in general the higher the temperature the higher the rate of chemical reaction. It is also recognized that moisture will not exist on the MPC exterior surfaces for many years since moisture will not condense on hot surfaces and the protection afforded by the HI-STORM FW overpack. It is estimated that it would take decades for the hottest MPC to approach ambient temperatures and once at ambient temperature, any MPC surfaces will be highly corrosion resistance even when wet.

#### 8.12.2.4 Cyclic Fatigue

As discussed in Section 3.1, 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 that is made from stainless alloy stock (endurance limit well in excess of 20,000 psi). All other off-normal or postulated accident conditions are infrequent or one-time occurrences, which cannot produce fatigue failures.

### 8.12.3 Compatibility of HI-STORM FW Overpack Materials

The principal operational considerations that bear on the adequacy of the storage overpack for the service life are addressed as follows:

#### Exposure to Environmental Effects

All exposed surfaces of the HI-STORM FW overpack are made from ferritic steels that are readily painted. Concrete, which serves strictly as a shielding material, is encased in steel. Therefore, the potential of environmental vagaries such as spalling of concrete, are ruled out for HI-STORM FW overpack. Under normal storage conditions, the bulk temperature of the HI-STORM FW overpack will change very gradually with time because of its large thermal inertia. Therefore, material degradation from rapid thermal ramping conditions is not credible for the HI-STORM FW overpack. Similarly, corrosion of structural steel embedded in the concrete structures due to salinity in the environment at coastal sites is not a concern for HI-STORM FW because HI-STORM FW does not rely on rebars (indeed, it contains no rebars). As discussed in Appendix 1.D of the HI-STORM 100 FSAR, the aggregates, cement and water used in the storage cask concrete are adequately controlled to provide high durability and resistance to temperature effects. The configuration of the storage overpack assures resistance to freeze-thaw degradation. In addition, the storage overpack is specifically designed for a full range of enveloping design basis natural phenomena that could occur over the service life of the storage overpack as catalogued in Section 2.2 and evaluated in Chapter 11.

### Material Degradation

The relatively low neutron flux to which the storage 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 ambient environment of the ISFSI storage pad mitigates damage due to exposure to corrosive and aggressive chemicals that may be produced at other industrial plants in the surrounding area.

### Maintenance and Inspection Provisions

The requirements for periodic inspection and maintenance of the storage overpack throughout its service life are defined in Section 10.2. These requirements include provisions for routine inspection of the storage overpack exterior and periodic visual verification that the ventilation flow paths of the storage overpack are free and clear of debris. ISFSIs located in areas subject to atmospheric conditions that 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-STORM FW system is designed for easy retrieval of the MPC from the storage overpack should it become necessary to perform more detailed inspections and repairs on the storage overpack.

The above findings are consistent with those of the NRC's Waste Confidence Decision Review [8.12.5], 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.

## 8.12.4 Compatibility of HI-TRAC VW Transfer Cask Materials

The principal design considerations that bear on the adequacy of the HI-TRAC VW Transfer Cask for the service life are addressed as follows:

### Exposure to Environmental Effects

All transfer cask materials that come in contact with the spent fuel pool are coated to facilitate decontamination. The HI-TRAC VW is designed for repeated normal condition handling operations with a high factor of safety to assure structural integrity. The resulting cyclic loading produces stresses that are well below the endurance limit of the cask's materials, and therefore, will not lead to a fatigue failure in the transfer cask. All other off-normal or postulated accident conditions are infrequent or one-time occurrences that do not contribute significantly to fatigue. In addition, the transfer cask utilizes materials that are not susceptible to brittle fracture during the lowest temperature permitted for loading, as discussed in Section 8.4 in the foregoing.

### Material Degradation

All transfer cask materials that are susceptible to corrosion are coated. The controlled environment in which the HI-TRAC VW is used mitigates damage due to direct exposure to corrosive chemicals that may be present in other industrial applications. The infrequent use and relatively low neutron flux to which the HI-TRAC VW materials are subjected do not result in radiation embrittlement or degradation of the shielding materials in the HI-TRAC VW that could impair the intended safety function. The HI-TRAC VW transfer cask materials have been selected for durability and wear resistance for their deployment.

### Maintenance and Inspection Provisions

The requirements for periodic inspection and maintenance of the HI-TRAC VW transfer cask throughout its service life are defined in Section 10.2. These requirements include provisions for routine inspection of the HI-TRAC VW transfer cask for damage prior to each use. Precautions are taken during bottom lid handling operations to protect the sealing surfaces of the bottom lid. The leak tightness of the liquid neutron shield is verified periodically. The water jacket pressure relief devices and connections for water injection/removal have been engineered for convenient removal and replacement.

#### 8.12.5 Potential Combustible Gas Generation

To ensure safe fuel loading operation the operating procedure described in Chapter 9 provides for the monitoring of hydrogen gas in the area around the MPC lid prior to and during welding or cutting activities. Although the aluminum surfaces (Metamic-HT basket and aluminum basket shims) are anodized, there is still a potential for generation of hydrogen in minute amounts when immersed in spent fuel pool water for an extended period. Accordingly, as a defense-in-depth measure, the lid welding procedure requires purging the space below the MPC lid prior to and during welding or cutting operation to eliminate any potential for formation of any combustible mixture of hydrogen and oxygen. Following the completion of the MPC lid welding and hydrostatic testing the MPC is drained and dried. As discussed earlier, after the completion of the drying operation there is no credible mechanism for any combustible gases to be generated within the MPC.

#### 8.12.6 Oxidation of Fuel During Loading/Unloading Operations

During the loading and unloading operations in a spent fuel pool, the fuel cladding is surrounded by water. During fuel drying operation the water is displaced with a non-oxidizing gas environment. Therefore, there is no credible mechanism for oxidation of fuel.

### 8.12.7 Conclusion

The above discussion leads to the conclusion that the materials selected for the HI-STORM FW System components are compatible with the environment for all operating conditions. There is no potential for significant corrosion, chemical reaction or galvanic reaction to shorten the intended service life of the equipment. In other words, the acceptance criteria set forth in ISG-15 are completely satisfied.

## 8.13 FUEL CLADDING INTEGRITY

### 8.13.1 Regulatory Guidance

The acceptance criteria from ISG-11 that apply to the fuel cladding are:

- a. For all fuel burnups (low and high), the maximum calculated fuel cladding temperature should not exceed 400°C (752°F) for normal conditions of storage and short-term loading operations (e.g., drying, backfilling with inert gas, and transfer of the cask to the storage pad).

However, for low burnup fuel, a higher short-term temperature limit may be used, if it can be shown by calculation that the best estimate cladding hoop stress is equal to or less than 90 MPa (13.053 psi) for the temperature limit proposed.

- b. During loading operations, for high burnup fuel, repeated thermal cycling (repeated heatup/cooldown cycles) may occur but should be limited to less than 10 cycles, with cladding temperature variations that are less than 65°C (149°F) each.
- c. For off-normal and accident conditions, the maximum cladding temperature should not exceed 570°C (1058°F).

The ISG-15 guidance on cladding integrity in its entirety provides the following supplemental requirements:

- a. The cladding temperature should be maintained below maximum allowable limits, and an inert environment should be maintained inside the cask cavity to maintain reasonable assurance that the spent fuel cladding will be protected against degradation that may lead to gross rupture, loss of retrievability, or severe degradation.
- b. Cladding should not rupture during re-flood operations.

### 8.13.2 Measures to Meet Regulatory Guidance

The HI-STORM FW System features and processes minimize the potential for any spent fuel cladding degradation during transfer and storage conditions by limiting the fuel cladding temperature and the environment around the fuel rod to within ISG-11 limits (Table 4.3.1).

The highly pure helium under positive pressure in the canister limits the amount of oxidants and controls the cladding temperature. The MPC drying and helium backfilling operations result in the creation of an inert environment around the fuel. As prescribed by NUREG-1536 [8.3.3], if the classical vacuum drying method is used, the partial pressure of water vapor is brought down to below 3 torr to minimize [8.13.1] residual oxidizing gas concentration.

An alternative method (preferred) for drying the MPC internals utilizes Holtec's patented Forced Helium Dehydration technology [8.13.1, 8.13.2] described in the HI-STORM 100 FSAR (Appendix 2.B). The Forced Helium Dehydrator has been successfully used at numerous nuclear plants since its regulatory approval in 2001. The efficacy of the Forced Gas Dehydrator (FGD) has been tested in a full-scale demonstration [8.13.4] for demineralizing simulated water-logged RBMK fuel [8.13.3].

The FHD uses helium as the working substance. The use of the FHD prevents the elevation of the fuel cladding temperature during drying, which is a chief demerit of the vacuum drying method. The use of the FHD method of drying is compulsory for high burnup fuel to protect its (relatively) ductility challenged cladding from severe thermal transients.

Chapter 2 provides the allowable fuel cladding temperature limits along with other design conditions. Chapter 4 presents performance evaluation of the HI-STORM FW System under normal conditions of storage, MPC temperatures during moisture removal operations and HI-STORM FW System long term storage maximum temperature conditions. Chapter 4 provides MPC temperatures under various accident conditions. It is demonstrated that the maximum calculated fuel cladding temperature is within 400°C (752°F) with substantial margins for normal conditions of storage and short-term loading operations. For off-normal and accident conditions, the maximum cladding temperature does not exceed 570°C (1058°F).

The short-term operations described in Chapter 9 are specifically configured to prevent severe thermal stresses in the fuel cladding due to rapid thermal transients.

The thermal stresses from MPC reflood analysis during fuel unloading operations shall be lower than typical MPCs because the HI-STORM FW fuel assemblies operate at considerably lower temperatures at Design Basis heat loads (see Chapter 4) than is permitted by ISG-11.

## 8.14 EXAMINATION AND TESTING

Examination and testing are integral parts of manufacturing of the HI-STORM FW System components. A comprehensive discussion on the examinations and testing that are conducted during the manufacturing process is provided in Section 10.1. The applicable codes and standards used are also referred and the acceptance criteria are listed.

### 8.14.1 Helium Leak Testing of Canister & Welds

Helium leakage testing of the MPC base metals (shell, baseplate, and MPC lid) and MPC shell to baseplate and shell to shell welds shall be performed in accordance with the leakage test methods and procedures of ANSI N14.5 [8.14.1]. Acceptance criterion is specified in Chapter 10. Testing shall be performed in accordance with written and approved procedures.

Leak testing results for the MPC shall be documented and shall become part of the quality record documentation package.

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 criteria are met.

Leakage testing of the field welded MPC lid-to-shell weld and closure ring welds are not required.

Leakage testing of the vent and drain port cover plate welds shall be performed after welding of the cover plates and subsequent NDE. The description and procedures for these field leakage tests are provided in Chapter 9 of this SAR and the acceptance criteria are defined in the Technical Specifications for the HI-STORM FW System.

### 8.14.2 Periodic Inspections

Post-fabrication inspections are discussed in Section 10.2 as part of the HI-STORM FW System maintenance program. Inspections are conducted prior to fuel loading or prior to each fuel handling campaign. Other periodic inspections are conducted during storage.

The HI-STORM FW overpack is a passive device with no moving parts. Overpack vent screens are inspected monthly for damage, holes, etc. The overpack external surface including identification markings is visually examined annually. The temperature monitoring system, if used, is inspected per licensee's QA program and manufacturer's recommendations. HI-TRAC VW transfer cask visual inspection is performed annually for compliance with the licensing drawings.



## 8.15 CONCLUSION

The preceding sections describe the materials used in important to safety SSCs and the suitability of those materials for their intended functions in the HI-STORM FW System.

The requirements of 10CFR72.122(a) are met: The material properties of SSCs important to safety conform to quality standards commensurate with their safety functions.

The requirements of 10CFR72.104(a), 106(b), 124, and 128(a)(2) are met: Materials used for criticality control and shielding are adequately designed and specified to perform their intended function.

The requirements of 10CFR72.122(h)(1) and 236(h) are met: The design of the DCSS and the selection of materials adequately protect the spent fuel cladding against degradation that might otherwise lead to gross rupture of the cladding.

The requirements of 10CFR72.236(h) and 236(m) are met: The material properties of SSCs important to safety will be maintained during normal, off-normal, and accident conditions of operation as well as short-term operations so the spent fuel or MPC, as appropriate, can be readily retrieved without posing operational safety problems.

The requirements of 10CFR72.236(g) are met: The material properties of SSCs important to safety will be maintained during all conditions of operation so the spent fuel can be safely stored for the specified service life and maintenance can be conducted as required.

The requirements of 10CFR72.236(h) are met: The HI-STORM FW System employs materials that are compatible with wet and dry spent fuel loading and unloading operations and facilities. These materials should not degrade over time or react with one another during long-term storage.

## 8.16 REFERENCES

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- [8.3.3] NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems,” U.S. Nuclear Regulatory Commission, Washington, DC, January 1997.
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- [8.12.3] Holtec Position Paper DS-330, “Estimating an Upper Bound on the Cumulative Corrosion in Stainless Steel MPCs in Highly Corrosive Environments.” (Holtec Proprietary)
- [8.12.4] Peckner and Bernstein, “Handbook of Stainless Steels,” First Ed., 1977.
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## APPENDIX 8.A

### Datasheets for Coatings and Paints<sup>§</sup>

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<sup>§</sup> The materials in this Appendix can also be found in the suppliers' website.



## Selection & Specification Data

Generic Type	Cycloaliphatic Amine Epoxy																		
Description	Highly chemical resistant epoxy mastic coating with exceptionally versatile uses in all industrial markets. Self-priming and suitable for application over most existing coatings, and tightly adherent to rust. Carboguard 890 serves as stand-alone system for a variety of chemical environments. Carboguard 890 is also designed for various immersion conditions.																		
Features	<ul style="list-style-type: none"><li>▪ Excellent chemical resistance</li><li>▪ Surface tolerant characteristics</li><li>▪ Conventional and low-temperature versions</li><li>▪ Self-priming and primer/finish capabilities</li><li>▪ Very good abrasion resistance</li><li>▪ VOC compliant to current AIM regulations</li><li>▪ Suitable for use in USDA inspected facilities</li></ul>																		
Color	Refer to Carboline Color Guide. Certain colors may require multiple coats for hiding. Note: The low temperature formulation will cause most colors to yellow or discolor more than normal in a short period of time. (Epoxies lose gloss, discolor and chalk in sunlight exposure.)																		
Finish	Gloss																		
Primers	Self-priming. May be applied over inorganic zinc primers and other tightly adhering coatings. A mist coat may be required to minimize bubbling over inorganic zinc primers.																		
Topcoats	Acrylics, Epoxies, Polyurethanes																		
Dry Film Thickness	4.0-6.0 mils (100-150 microns) per coat 6.0-8.0 mils (150-200 microns) over light rust and for uniform gloss over inorganic zincs. Don't exceed 10 mils (250 microns) in a single coat. Excessive film thickness over inorganic zincs may increase damage during shipping or erection.																		
Solids Content	By Volume (890): 75% ± 2% (890LT): 80% ± 2%																		
Theoretical Coverage Rate	890: 1203 mil ft <sup>2</sup> (30.0 m <sup>2</sup> /l at 25 microns) 241 ft <sup>2</sup> at 5 mils (6.0 m <sup>2</sup> /l at 125 microns) 890LT: 1283 mil ft <sup>2</sup> (31.0 m <sup>2</sup> /l at 25 microns) 257 ft <sup>2</sup> at 5 mils (6.3 m <sup>2</sup> /l at 125 microns) Allow for loss in mixing and application																		
VOC Values	<table><tr><td></td><td><b>890</b></td><td><b>890 LT</b></td></tr><tr><td>As supplied</td><td>1.7lbs/gal (214 g/l)</td><td>1.5lbs/gal (180g/l)</td></tr><tr><td>Thinned w/#2*: (250g/l)</td><td>7oz/gal=2.0lbs/gal (250g/l)</td><td>15oz/gal=2.0lbs/gal (250g/l)</td></tr><tr><td></td><td>13oz/gal=2.2lbs/gal (271g/l)</td><td></td></tr><tr><td>Thinned w/#33*: (250g/l)</td><td>7oz/gal=2.0lbs/gal (250g/l)</td><td>14oz/gal=2.0 lbs/gal (250g/l)</td></tr><tr><td></td><td>16oz/gal=2.3lbs/gal (285g/l)</td><td>16oz/gal=2.1lbs/gal (258g/l)</td></tr></table> <p>*Use Thinner #76 up to 8 oz/gal for 890 and 16 oz/gal for 890 LT where non-photochemically reactive solvents are required.</p>		<b>890</b>	<b>890 LT</b>	As supplied	1.7lbs/gal (214 g/l)	1.5lbs/gal (180g/l)	Thinned w/#2*: (250g/l)	7oz/gal=2.0lbs/gal (250g/l)	15oz/gal=2.0lbs/gal (250g/l)		13oz/gal=2.2lbs/gal (271g/l)		Thinned w/#33*: (250g/l)	7oz/gal=2.0lbs/gal (250g/l)	14oz/gal=2.0 lbs/gal (250g/l)		16oz/gal=2.3lbs/gal (285g/l)	16oz/gal=2.1lbs/gal (258g/l)
	<b>890</b>	<b>890 LT</b>																	
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Thinned w/#33*: (250g/l)	7oz/gal=2.0lbs/gal (250g/l)	14oz/gal=2.0 lbs/gal (250g/l)																	
	16oz/gal=2.3lbs/gal (285g/l)	16oz/gal=2.1lbs/gal (258g/l)																	
Dry Temp. Resistance	Continuous: 250°F (121°C) Non-Continuous: 300°F (149°C) Discoloration and loss of gloss is observed above 200°F (93°C).																		

**Limitations** Do not apply over latex coatings. For immersion April 2007 replaces February 2007

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projects use only factory made material in special colors. Consult Technical Service for specifics. Carboguard 890 LT should not be used for immersion and should only be used as a primer or intermediate coat. Discoloration may be objectionable if used as a topcoat.

## Substrates & Surface Preparation

<b>General</b>	Surfaces must be clean and dry. Employ adequate methods to remove dirt, dust, oil and all other contaminants that could interfere with adhesion of the coating.
<b>Steel</b>	Immersion: SSPC-SP10 Non-immersion: SSPC-SP6 1.5-3.0 mils (38-75 microns) SSPC-SP2 or SP3 are suitable cleaning methods for mild environments.
<b>Galvanized Steel</b>	Prime with specific Carboline primers as recommended by your Carboline Sales Representative. Refer to the specific primer's Product Data Sheet for substrate preparation requirements.
<b>Concrete or CMU</b>	Concrete must be cured 28 days at 75°F (24°C) and 50% relative humidity or equivalent. Prepare surfaces in accordance with ASTM D4258 Surface Cleaning of Concrete and ASTM D4259 Abrading Concrete. Voids in concrete may require surfacing. Mortar joints should be cured a min of 15 days. Prime with itself, Carboguard® 1340, or suitable filler/sealer.
<b>Drywall &amp; Plaster</b>	Joint compound and plaster should be fully cured prior to coating application. Prime with Carbocrylic® 120 or Carboguard 1340.
<b>Previously Painted Surfaces</b>	Lightly sand or abrade to roughen surface and degloss the surface. Existing paint must attain a minimum 3B rating in accordance with ASTM D3359 "X-Scratch" adhesion test.

## Performance Data

Test Method	System	Results	Report #
ASTM D3359 Adhesion	Blasted Steel 1 ct. 890	5A	0270
ASTM D4060 Abrasion	Blasted Steel 1 ct. Epoxy Pr. 1 ct. 890	85 mg. loss after 1000 cycles, CS17 wheel, 1000 gm. load	02411
ASTM B117 Salt Fog	Blasted Steel 2 cts. 890	No effect on plane, rust in scribe, 1/16" undercutting at scribe after 2000 hours	02594
ASTM B117 Salt Fog	Blasted Steel 1 ct. IOZ 1 ct. 890	No effect on plane, no rust in scribe and no undercutting after 4000 hours	L40-42,45,95
ASTM D1735 Water Fog	Blasted Steel 1 ct. Epoxy Pr. 1 ct. 890	No blistering, rusting or delamination after 2800 hours	08564
ASTM D3363 Pencil Hardness	Blasted Steel 2 cts. 890	Greater than 8H	02775
ASTM D2486 Scrub Resistance	Blasted Steel 1 ct. 890	93% gloss retained after 10,000 cycles w/ liquid scrub medium	03142
ASTM E84 Flame and Smoke	2 ct. 890	5 Flame 5 Smoke Class A	03110

Test reports and additional data available upon written request.

0986/0983

## Carboguard® 890 & 890 LT

### Application Equipment

Listed below are general equipment guidelines for the application of this product. Job site conditions may require modifications to these guidelines to achieve the desired results. **General Guidelines:**

<b>Spray Application (General)</b>	This is a high solids coating and may require adjustments in spray techniques. Wet film thickness is easily and quickly achieved. The following spray equipment has been found suitable and is available from manufacturers such as Binks, DeVilbiss and Graco.
<b>Conventional Spray</b>	Pressure pot equipped with dual regulators, 3/8" I.D. minimum material hose, .070" I.D. fluid tip and appropriate air cap.
<b>Airless Spray</b>	Pump Ratio: 30:1 (min.)* GPM Output: 3.0 (min.) Material Hose: 3/8" I.D. (min.) Tip Size: .017"-.021" Output PSI: 2100-2300 Filter Size: 60 mesh *Teflon packings are recommended and available from the pump manufacturer.
<b>Brush &amp; Roller (General)</b>	Multiple coats may be required to obtain desired appearance, recommended dry film thickness and adequate hiding. Avoid excessive re-brushing or re-rolling. For best results, tie-in within 10 minutes at 75°F (24°C).
<b>Brush</b>	Use a medium bristle brush.
<b>Roller</b>	Use a short-nap synthetic roller cover with phenolic core.

### Mixing & Thinning

<b>Mixing</b>	Power mix separately, then combine and power mix. DO NOT MIX PARTIAL KITS.
<b>Ratio</b>	890 and 890 LT 1:1 Ratio (A to B)
<b>Thinning*</b>	Spray: Up to 13 oz/gal (10%) w/ #2 Brush: Up to 16 oz/gal (12%) w/ #33 Roller: Up to 16 oz/gal (12%) w/ #33 Thinner #33 can be used for spray in hot/windy conditions. Use of thinners other than those supplied or recommended by Carboline may adversely affect product performance and void product warranty, whether expressed or implied. *See VOC values for thinning limits.
<b>Pot Life</b>	890 3 Hours at 75°F (24°C) 890 LT 2 Hours at 75°F (24°C) Pot life ends when coating loses body and begins to sag. Pot life times will be less at higher temperatures.

### Cleanup & Safety

<b>Cleanup</b>	Use Thinner #2 or Acetone. In case of spillage, absorb and dispose of in accordance with local applicable regulations.
<b>Safety</b>	Read and follow all caution statements on this product data sheet and on the MSDS for this product. Employ normal workmanlike safety precautions. Hypersensitive persons should wear protective clothing, gloves and use protective cream on face, hands and all exposed areas.
<b>Ventilation</b>	When used as a tank lining or in enclosed areas, thorough air circulation must be used during and after application until the coating is cured. The ventilation system should be capable of preventing the solvent vapor concentration from reaching the lower explosion limit for the solvents used. User should test and monitor exposure levels to insure all personnel are below guidelines. If not sure or if not able to monitor levels, use MSHA/NIOSH approved supplied air respirator.
<b>Caution</b>	This product contains flammable solvents. Keep away from sparks and open flames. All electrical equipment and installations should be made and grounded in accordance with the National Electric Code. In areas where explosion hazards exist, workmen should be required to use non-ferrous tools and wear conductive and non-sparking shoes.

April 2007 replaces February 2007

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### Application Conditions

890				
Condition	Material	Surface	Ambient	Humidity
Normal	60°-85°F (16°-29°C)	60°-85°F (16°-29°C)	60°-90°F (16°-32°C)	0-80%
Minimum	50°F (10°C)	50°F (10°C)	50°F (10°C)	0%
Maximum	90°F (32°C)	125°F (52°C)	110°F (43°C)	80%

890 LT				
Condition	Material	Surface	Ambient	Humidity
Normal	60-85°F (16-29°C)	60-85°F (16-29°C)	60-90°F (16-32°C)	10-80%
Minimum	40°F (4°C)	35°F (2°C)	35°F (2°C)	0%
Maximum	90°F (32°C)	125°F (52°C)	110°F (43°C)	80%

This product simply requires the substrate temperature to be above the dew point. Condensation due to substrate temperatures below the dew point can cause flash rusting on prepared steel and interfere with proper adhesion to the substrate. Special application techniques may be required above or below normal application conditions.

### Curing Schedule

890 (Based on 4-8 mils, 100-200 microns dry film thickness.)				
Surface Temp. & 50% Relative Humidity	Dry to Recoat	Dry to Topcoat w/ Other Finishes	Final Cure	
			General	Immersion
50°F (10°C)	12 Hours	24 Hours	3 Days	N/R
60°F (16°C)	8 Hours	16 Hours	2 Days	10 Days
75°F (24°C)	4 Hours	8 Hours	1 Day	5 Days
90°F (32°C)	2 Hours	4 Hours	16 Hours	3 Days

890 LT (Based on 5 mils, 125 microns dry film thickness.)				
Surface Temp. & 50% Relative Humidity	Dry to Touch	Dry to Handle	Dry to Recoat & Topcoat w/ Others	Final Cure General Service
35°F (2°C)	5 Hours	18 Hours	20 Hours	7 Days
40°F (4°C)	4.5 Hours	15.5 Hours	16 Hours	5 Days
50°F (10°C)	3.5 Hours	6.5 Hours	12 Hours	3 Days
60°F (16°C)	2 Hours	5 Hours	8 Hours	2 Days
75°F (24°C)	1.5 Hours	2 Hours	4 Hours	24 Hours
90°F (32°C)	1 Hour	1.5 Hours	2 Hours	16 Hours

Higher film thickness, insufficient ventilation or cooler temperatures will require longer cure times and could result in solvent entrapment and premature failure. Excessive humidity or condensation on the surface during curing can interfere with the cure, can cause discoloration and may result in a surface haze. Any haze or blush must be removed by water washing before recoating. During high humidity conditions, it is recommended that the application be done while temperatures are increasing. **Maximum recoat/topcoat times are 30 days for epoxies and 90 days for polyurethanes at 75°F (24°C).** If the maximum recoat times have been exceeded, the surface must be abraded by sweep blasting or sanding prior to the application of additional coats. 890 LT applied below 50°F (10°C) may temporarily soften as temperatures rise to 60°F (16°C). This is a normal condition and will not affect performance.

### Packaging, Handling & Storage

<b>Shipping Weight (Approximate)</b>	<b>2 Gallon Kit</b> 29 lbs (13 kg)	<b>10 Gallon Kit</b> 145 lbs (66 kg)
<b>Flash Point (Setflash)</b>	89°F (32°C) for Part A; 890 & 890 LT 73°F (23°C) for Part B; 890 & 890 LT	
<b>Storage Temperature &amp; Humidity</b>	40°-110°F (4°-43°C) Store indoors. 0-100% Relative Humidity	
<b>Shelf Life: 890 &amp; 890 LT</b>	Part A: Min. 36 months at 75°F (24°C) 890 Part B: Min. 15 months at 75°F (24°C) 890 LT Part B: Min. 15 months at 75°F (24°C)	

\*Shelf Life: (actual stated shelf life) when kept at recommended storage conditions and in original unopened containers.

**Carboline.**

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## Selection & Specification Data

Generic Type	Solvent Based Inorganic Zinc					
Description	Time-tested corrosion resistant primer that protects steel galvanically in the harshest environments. For over four decades, Carbozinc 11 (CZ 11) has been the industry standard for high-performance inorganic zinc protection on steel structures worldwide.					
Features	<ul style="list-style-type: none"><li>▪ CZ 11 and CZ 11 FG meet Class B slip co-efficient and creep testing criteria for use on faying surfaces</li><li>▪ Rapid cure. Dry to handle in 45 minutes at 60°F (16°C) and 50% relative humidity.</li><li>▪ Low temperature cure down to 0°F (-18°C).</li><li>▪ High zinc loading.</li><li>▪ Meets FDA requirements in gray color.</li><li>▪ Available in ASTM D520, Type II zinc version.</li><li>▪ Very good resistance to salting.</li><li>▪ May be applied with standard airless or conventional spray equipment.</li><li>▪ VOC compliant in certain areas</li></ul>					
CZ 11 FG	<ul style="list-style-type: none"><li>▪ Lower zinc loading for economics.</li><li>▪ VOC compliant for shop/fabricator use only.</li></ul>					
Color	Green (0300); Gray (0700)					
Finish	Flat					
Primers	Self Priming					
Topcoats	Not required for certain exposures. Can be topcoated with Epoxies, Polyurethanes, Acrylics, High-Heat Silicones and others as recommended by your Carboline sales representative. Under certain conditions, a mist coat is required to minimize topcoat bubbling.					
Dry Film Thickness	2.0-3.0 mils (50-75 microns). Dry film thickness in excess of 6.0 mils (150 microns) per coat is not recommended.					
Solids Content	By Weight:	<table><tr><td><u>CZ 11</u></td><td><u>CZ 11 FG</u></td></tr><tr><td>79% ± 2%</td><td>74% ± 2%</td></tr></table>	<u>CZ 11</u>	<u>CZ 11 FG</u>	79% ± 2%	74% ± 2%
<u>CZ 11</u>	<u>CZ 11 FG</u>					
79% ± 2%	74% ± 2%					
Zinc Content In dry film	By Weight:	<table><tr><td><u>CZ 11</u></td><td><u>CZ 11 FG</u></td></tr><tr><td>85% ± 2%</td><td>79% ± 2%</td></tr></table>	<u>CZ 11</u>	<u>CZ 11 FG</u>	85% ± 2%	79% ± 2%
<u>CZ 11</u>	<u>CZ 11 FG</u>					
85% ± 2%	79% ± 2%					
Theoretical Coverage Rate	CZ 11: 1000 mil ft <sup>2</sup> (22.8 m <sup>2</sup> /l at 25 microns) 333 ft <sup>2</sup> at 3.0 mils (8.2 m <sup>2</sup> /l at 75 microns) CZ 11 FG: 850 mil ft <sup>2</sup> (19.4 m <sup>2</sup> /l at 25 microns) 283 ft <sup>2</sup> at 3.0 mils (7.0 m <sup>2</sup> /l at 75 microns) Allow for loss in mixing and application					
VOC Values Carbozinc 11	EPA Method 24: 4.0 lbs./gal (479 g/l) Thinned: 7 oz/gal w/ #21: 4.1 lbs./gal (492 g/l) 5 oz/gal w/ #26: 4.1 lbs./gal (492 g/l) 5 oz/gal w/ #33: 4.1 lbs./gal (492 g/l) These are nominal values.					
VOC Values Carbozinc 11 FG	EPA Method 24: 4.3 lbs./gal (515 g/l) Thinned: For use in fabrication shops only to remain in VOC compliance in accordance with EPA Standards. 7 oz/gal w/ #21: 4.5 lbs./gal ( 539 g/l) 5 oz/gal w/ #26: 4.5 lbs./gal ( 539 g/l) 5 oz/gal w/ #33: 4.5 lbs./gal ( 539 g/l) These are nominal values.					
Dry Temp. Resistance	<u>Untopcoated:</u> Continuous: 750°F (399°C) Non-Continuous: 800°F (427°C) <u>With recommended silicone topcoats:</u> Continuous: 1000°F (538°C) Non-Continuous: 1200°F (649°C)					

## Substrates & Surface Preparation

<b>General</b>	Surfaces must be clean and dry. Employ adequate methods to remove dirt, dust, oil and all other contaminants that could interfere with adhesion of the coating.	
<b>Steel</b>	<u>Non-Immersion:</u>	SSPC-SP6 and obtain a 1.0-3.0 mil (25-75 micron) angular blast profile.

## Performance Data

CZ 11			
Test Method	System	Results	Report #
ASTM A-325 Slip Co-efficient	Blasted steel 1 ct. CZ 11	0.668; meets requirements for Class B rating	02722
ASTM B117 Salt Spray	1 ct. CZ 11 at 2 mils dry film thickness over blasted steel	No rusting or blistering, cracking or delamination after 43000 hrs. Moderate salting of the surface only.	SR 408
ASTM D3363 Pencil Hardness	1 ct. CZ 11	Pencil Hardness "2H"	03278
AASHTO M300 Bullet Hole Immersion Paragraph 4.6.9	1 ct. CZ 11 over Abrasive blasted steel	No blistering or rusting of coating or rusting of bare steel area after 650 hrs. Immersion in 5% sodium chloride solution; 1.5" round bare area in coating.	02514

Test reports and additional data available upon written request.

## Application Equipment

Listed below are general equipment guidelines for the application of this product. Job site conditions may require modification to these guidelines to achieve the desired results.

### General Guidelines:

<b>Spray Application (General)</b>	The following spray equipment has been found suitable and is available from manufacturers such as Binks, DeVilbiss and Graco. Keep material under mild agitation during application. If spraying stops for more than 10 minutes, recirculate the material remaining in the spray line. Do not leave mixed primer in the hoses during work stoppages.
<b>Conventional Spray</b>	Agitated pressure pot equipped with dual regulators, 3/8" I.D. minimum material hose, with a maximum length of 50', .070" I.D. fluid tip and appropriate air cap.
<b>Airless Spray</b>	Pump Ratio: 30:1 (min.) GPM Output: 3.0 (min.) Material Hose: 3/8" I.D. (min.) Tip Size: .019-.023" Output PSI: 1500-2000 Filter Size: 60 mesh Teflon packings are recommended and available from the pump manufacturer.
<b>Brush</b>	For touch-up of areas less than one square foot only. Use medium bristle brush and avoid rebrushing.
<b>Roller</b>	Not recommended

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0250

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## Carbozinc® 11

### Mixing & Thinning

<b>Mixing</b>	Power mix base, then combine and power mix as follows. Pour zinc filler very slowly into premixed base with continuous agitation. Mix until free of lumps. Pour mixture through a 30 mesh screen. DO NOT MIX PARTIAL KITS. Tip: Sifting zinc through a window screen will aid in the mixing process by breaking up or catching dry zinc lumps.		
<b>Ratio</b>	<u>CZ 11</u> <b>1 Gal Kit</b> Part A: .75 gal. Zinc Filler: 14.6 lbs.	<u>CZ 11</u> <b>5 Gallon Kit</b> 3.75 gallons 73 lbs.	<u>CZ 11 FG</u> <b>4.6 Gallon Kit</b> 3.75 gallons 50 lbs.
<b>Thinning</b>	May be thinned up to 5 oz/gal (4%) with #26 for ambient and warm surfaces. For extremely warm or windy conditions, may be thinned up to 5 oz/gal (4%) with #33. In cool weather (below 40°F (4°C)), thin up to 7 oz/gal (6%) with #21. Use of thinners other than those supplied or recommended by Carboline may adversely affect product performance and void product warranty, whether expressed or implied.		
<b>Pot Life</b>	8 Hours at 75°F (24°C) and less at higher temperatures. Pot life ends when coating becomes too viscous to use.		

### Cleanup & Safety

<b>Cleanup</b>	Use Thinner #21 or Isopropyl Alcohol. In case of spillage, absorb and dispose of in accordance with local applicable regulations.
<b>Safety</b>	Read and follow all caution statements on this product data sheet and on the MSDS for this product. Employ normal workmanlike safety precautions. Hypersensitive persons should wear protective clothing, gloves and use protective cream on face, hands and all exposed areas.
<b>Ventilation</b>	When used as a tank lining or in enclosed areas, thorough air circulation must be used during and after application until the coating is cured. The ventilation system should be capable of preventing the solvent vapor concentration from reaching the lower explosion limit for the solvents used. In addition to ensuring proper ventilation, appropriate respirators must be used by all application personnel.
<b>Caution</b>	This product contains flammable solvents. Keep away from sparks and open flames. All electrical equipment and installations should be made and grounded in accordance with the National Electric Code. In areas where explosion hazards exist, workmen should be required to use non-ferrous tools and wear conductive and non-sparking shoes.

### Application Conditions

Condition	Material	Surface	Ambient	Humidity
Normal	40°-95°F (4°-35°C)	40°-110°F (4°-43°C)	40°-95°F (4°-35°C)	40-90%
Minimum	0°F (-18°C)	0°F (-18°C)	0°F (-18°C)	30%
Maximum	130°F (54°C)	200°F (93°C)	130°F (54°C)	95%

This product simply requires the substrate temperature to be above the dew point. Condensation due to substrate temperatures below the dew point can cause flash rusting on prepared steel and interfere with proper adhesion to the substrate. Special application techniques may be required above or below normal application conditions.

### Curing Schedule

Surface Temp. & 50% Relative Humidity	Dry to Handle	Dry to Topcoat/Recoat
0°F (-18°C)	4 Hours	7 Days
40°F (4°C)	1 Hour	48 Hours
60°F (16°C)	¾ Hour	24 Hours
80°F (27°C)	¾ Hour	18 Hours
100°F (38°C)	¾ Hour	16 Hours

These times are based on a 3.0-4.0 mil (75-100 micron) dry film thickness. Higher film thickness, insufficient ventilation or cooler temperatures will require longer cure times and could result in solvent entrapment and premature failure. Humidity levels below 50% will require longer cure times. **Notes:** Any salting that appears on the zinc surface as a result of prolonged weathering exposure must be removed prior to the application of additional coatings. Also, loose zinc must be removed from the cured film by rubbing with fiberglass screen wire if: 1) The Carbozinc 11 is to be used without a topcoat in immersion service and "zinc pick up" could be detrimental, or 2) When "dry spray/overspray" is evident on the cured film and a topcoat will be applied. For **accelerated curing** or where the **relative humidity is below 40%**, allow an initial 2-hour ambient cure. Follow 2 hour cure with water misting or steam to keep the coated surface wet for a minimum of 8 hours and until the coated surface achieves a "2H" pencil hardness per ASTM D3363.

### Packaging, Handling & Storage

<b>CZ 11 Shipping Weight (Approximate)</b>	<u>1 Gallon Kit</u> 23 lbs (10 kg)	<u>5 Gallon Kit</u> 113 lbs (51 kg)
<b>CZ 11 FG Shipping Weight (Approximate)</b>	<u>4.6 Gallon Kit</u> 104 lbs. (47 kg)	
<b>Flash Point (Setflash)</b>	Part A: 55°F (13°C) Zinc Filler: NA	
<b>Storage (General)</b>	Store Indoors.	
<b>Storage Temperature &amp; Humidity</b>	40° -100°F (4-38°C). 0-90% Relative Humidity	
<b>Shelf Life: 11 &amp; 11FG</b>	Part A: 12 months at 75°F (24°C) Part B: 24 months at 75°F (24°C)	
<b>*Shelf Life: (actual stated shelf life) when kept at recommended storage conditions and in original unopened containers.</b>		



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## Selection & Specification Data

<b>Generic Type</b>	Amine-Cured Novolac Epoxy
<b>Description</b>	Highly cross-linked, glass flake-filled polymer that offers exceptional barrier protection and resistance to wet/dry cycling at elevated temperatures. Suitable for insulated and non-insulated pipes, stacks and equipment operating up to 450°F (232°C). This coating provides excellent resistance to corrosion, abrasion and permeation, and its novolac-modification resists severe chemical attack.
<b>Features</b>	<ul style="list-style-type: none"> <li>Temperature resistance up to 450°F (232°C)</li> <li>High-build single-coat capabilities</li> <li>Excellent resistance to thermal shock</li> <li>Superior abrasion and chemical resistance through internal reinforcement</li> <li>Ambient-temperature cure</li> <li>VOC compliant to current AIM regulations</li> </ul>
<b>Color</b>	Red (0500); Gray (5742)
<b>Finish</b>	Eggshell
<b>Primers</b>	Self-priming. May be applied over epoxies and phenolics.
<b>Topcoats</b>	Epoxies, Polyurethanes
<b>Dry Film Thickness</b>	8.0-10.0 mils (200-250 microns) Do not exceed 15 mils (375 microns) per coat.
<b>Solids Content</b>	By Volume: 70% ± 2%
<b>Theoretical Coverage Rate</b>	1117 mil ft <sup>2</sup> (27.9 m <sup>2</sup> /l at 25 microns) Allow for loss in mixing and application
<b>VOC Values</b>	As supplied: 2.08 lbs/gal (250 g/l) Thinned: 13 oz/gal w/ #213: 2.58 lbs/gal (308 g/l) 13 oz/gal w/#2 2.54 lbs/gal (305 g/l) These are nominal values.
<b>Dry Temp. Resistance</b>	Continuous: 425°F (218°C) Non-Continuous: 450°F (232°C) Discoloration and loss of gloss may be observed above 200°F (93°C).
<b>Limitations</b>	Epoxies lose gloss, discolor and eventually chalk in sunlight exposure.

## Substrates & Surface Preparation

<b>General</b>	Surfaces must be clean and dry. Employ adequate methods to remove dirt, dust, oil and all other contaminants that could interfere with adhesion of the coating.	
<b>Steel</b>	<u>Non-Insulated:</u>	SSPC-SP6
	<u>Insulated:</u>	SSPC-SP10
	<u>Surface Profile:</u>	2.0-3.0 mils (50-75 microns)

## Performance Data

Test Method	System	Results	Report #
ASTM D3359 Adhesion	Blasted Steel 2 cts. 450	4A	08460
ASTM D4060 Abrasion	Blasted Steel 2 cts. 450	171 mg loss after 1000 cycles; CS17 wheel, 1000 gram load	02910
ASTM D2794 Impact	Blasted Steel 1 ct. 450	.375 in. from damaged area, 100-in./lbs	02675
Heat Cycling Test	Blasted Steel 1 ct. 450	No cracking, blistering or delamination of film after 425°F for 1 hr/ambient/ -10°F for 24 hrs/ambient/ 425°F for 24 hrs/ambient/ -10°F for 24 hrs/ambient/ 425°F for 200 hr/ambient	SR342
Modified NACE Std. TM-01-74B Immersion	Blasted Steel 2 cts. 450	No effect to coating film except discoloration after 6 month exposure, Deionized water	02551
Chemical Resistance	Blasted Steel 1 ct. 450	Resistant to fumes of commons acids, alkalis, solvents and hydrocarbon compounds. Resistant to splash and spillage of alkalis, solvents and hydrocarbons. Acid contact may cause discoloration of coating.	SR 359 02735 03133 02794

Test reports and additional data available upon written request.

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## Thermaline® 450 Novolac

### Application Equipment

Listed below are general equipment guidelines for the application of this product. Job site conditions may require modifications to these guidelines to achieve the desired results.

#### General Guidelines:

<b>Spray Application (General)</b>	The following spray equipment has been found suitable and is available from manufacturers such as Binks, DeVilbiss and Graco.
<b>Conventional Spray</b>	Pressure pot equipped with dual regulators, ½" I.D. minimum material hose, .110" I.D. fluid tip and appropriate air cap.
<b>Airless Spray</b>	Pump Ratio: 45:1 (min.) GPM Output: 3.0 (min.) Material Hose: ½" I.D. (min.) Tip Size: .035-.041" Output PSI: 2200-2500 *Teflon packings are recommended and available from the pump manufacturer.
<b>Brush</b>	For striping of welds and touch-up of small areas only. Use a medium natural bristle brush and avoid rebrushing.
<b>Roller</b>	Not recommended.

### Mixing & Thinning

<b>Mixing</b>	Power mix separately, then combine and power mix. DO NOT MIX PARTIAL KITS.
<b>Ratio</b>	4:1 Ratio (A to B)
<b>Thinning</b>	May be thinned up to 13 oz/gal (10%) with Thinner #213. For application on horizontal surfaces, may be thinned up to 13 oz/gal (10%) with Thinner #2. Agitate Thinner #213 before use. Thinner #213 will have a thick viscous appearance which is normal. Use of thinners other than those supplied by Carboline may adversely affect product performance and void product warranty, whether expressed or implied.
<b>Pot Life</b>	3 Hours at 75°F (24°C). Pot life ends when coating loses body and begins to sag. Pot life times will be less at higher temperatures.

### Cleanup & Safety

<b>Cleanup</b>	Use Thinner #2 or Acetone. In case of spillage, absorb and dispose of in accordance with local applicable regulations.
<b>Safety</b>	Read and follow all caution statements on this product data sheet and on the MSDS for this product. Employ normal workmanlike safety precautions. Hypersensitive persons should wear protective clothing, gloves and use protective cream on face, hands and all exposed areas.
<b>Ventilation</b>	When used in enclosed areas, thorough air circulation must be used during and after application until the coating is cured. The ventilation system should be capable of preventing the solvent vapor concentration from reaching the lower explosion limit for the solvents used. User should test and monitor exposure levels to insure all personnel are below guidelines. If not sure or if not able to monitor levels, use MSHA/NIOSH approved supplied air respirator.
<b>Caution</b>	This product contains flammable solvents. Keep away from sparks and open flames. All electrical equipment and installations should be made and grounded in accordance with the National Electric Code. In areas where explosion hazards exist, workmen should be required to use non-ferrous tools and wear conductive and non-sparking shoes.

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### Application Conditions

Condition	Material	Surface	Ambient	Humidity
Normal	65°-85°F (18°-29°C)	65°-85°F (18°-29°C)	65°-85°F (18°-29°C)	30-60%
Minimum	55°F (13°C)	50°F (10°C)	50°F (10°C)	0%
Maximum	90°F (32°C)	110°F (43°C)	100°F (38°C)	85%

This product simply requires the substrate temperature to be above the dew point. Condensation due to substrate temperatures below the dew point can cause flash rusting on prepared steel and interfere with proper adhesion to the substrate. Special application techniques may be required above or below normal application conditions.

### Curing Schedule

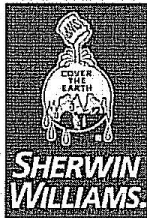
Surface Temp. & 50% Relative Humidity	Dry to Handle	Dry to Topcoat w/ Other Finishes	Final Cure
50°F (10°C)	18 Hours	48 Hours	21 Days
60°F (16°C)	12 Hours	32 Hours	14 Days
75°F (24°C)	6 Hours	16 Hours	7 Days
90°F (32°C)	3 Hours	8 Hours	4 Days

These times are based on a 10.0 mil (250 micron) dry film thickness. Higher film thickness, insufficient ventilation or cooler temperatures will require longer cure times and could result in solvent entrapment and premature failure. Excessive humidity or condensation on the surface during curing can interfere with the cure, can cause discoloration and may result in a surface haze. Any haze or blush must be removed by water washing before recoating. During high humidity conditions, it is recommended that the application be done while temperatures are increasing. If the final cure time is exceeded, the surface must be abraded by sweep blasting prior to the application of additional coats.

### Packaging, Handling & Storage

Shipping Weight (Approximate)	<u>1 Gallon Kit</u> 12 lbs (6 kg)	<u>5 Gallon Kit</u> 58 lbs (26 kg)
Flash Point (Setflash)	Part A: 53°F (12°C) Part B: >200°F (93°C)	
Storage (General)	Store Indoors.	
Storage Temperature & Humidity	40° - 110°F (4°-43°C) 0-90% Relative Humidity	
Shelf Life	Part A & B: Min. 36 months at 75°F (24°C)	
<b>*Shelf Life: (actual stated shelf life) when kept at recommended storage conditions and in original unopened containers.</b>		





**Industrial  
&  
Marine  
Coatings**

## ZINC CLAD® II PLUS INORGANIC ZINC-RICH COATING

PART A	B69VZ12	BASE
PART B	B69VZ13	ACCELERATOR
PART C	B69VZ15	ACCELERATOR
PART F	B69D11	ZINC DUST

6.13

PRODUCT INFORMATION			Revised 12/05																							
PRODUCT DESCRIPTION		RECOMMENDED USES																								
<p><b>ZINC CLAD II PLUS</b> is a solvent-based, three component, inorganic ethyl silicate, zinc rich coating. This is fast drying, high solids, low VOC coating with 82%, by weight, of zinc dust in the dry film.</p> <ul style="list-style-type: none"><li>• Coating self-heals to resume protection if damaged</li><li>• Provides cathodic/sacrificial protection by the same mechanism as galvanizing</li><li>• Forms an inorganic barrier to moisture and solvents</li><li>• Meets Class B requirements for Slip Coefficient and Creep Resistance, 0.67</li><li>• Meets AASHTO M-300 specification</li></ul>		<p>For use over prepared blasted steel and galvanized steel in areas such as:</p> <ul style="list-style-type: none"><li>• Bridges</li><li>• Shop or field application</li><li>• As a one-coat maintenance coating or as a permanent primer for severe corrosive environments (pH range 5-9)</li><li>• Ideal for application at low temperatures or service at high temperatures and/or humidity conditions</li><li>• Fresh and demineralized water immersion service (non-potable)</li><li>• Compliance with Class B Slip Coefficient rating when used alone or as part of a system with Steel Spec Epoxy Primer as a topcoat</li></ul>																								
PRODUCT CHARACTERISTICS		PERFORMANCE CHARACTERISTICS																								
<p><b>Finish:</b></p> <p><b>Color:</b></p> <p><b>Volume Solid:</b></p> <p><b>Weight Solid:</b></p> <p><b>VOC (EPA Method 24):</b></p> <p><b>Zinc Content in Dry Film:</b></p> <p><b>Mix Ratio:</b></p> <p><b>Recommended Spreading Rate per coat:</b></p> <p>Wet mils: 3.0 - 6.0 Dry mils: 2.0 - 4.0 Coverage: 400 - 610 sq ft/gal approximate</p> <p><b>Note:</b> Brush application is for small areas only. Application of coating above maximum or below minimum recommended spreading rate may adversely affect coating performance.</p> <p><b>Drying Schedule @ 4.0 mils wet @ 50% RH:</b></p> <table><tr><td></td><td>@ 40°F</td><td>@ 77°F</td><td>@ 100°F</td></tr><tr><td>To touch:</td><td>25 minutes</td><td>20 minutes</td><td>5 minutes</td></tr><tr><td>To handle:</td><td>1 hour</td><td>20 minutes</td><td>15 minutes</td></tr><tr><td>To topcoat:</td><td>7 days</td><td>24 hours</td><td>8 hours</td></tr><tr><td>To cure:</td><td>7 days</td><td>36 hours</td><td>24 hours</td></tr><tr><td>To stack</td><td>6 hours</td><td>2 hours</td><td>1 hour</td></tr></table> <p>Drying time is temperature, humidity, and film thickness dependent.</p> <p><b>Pot Life:</b></p> <p>8 hours @ 77°F High humidity will shorten pot life</p> <p><b>Sweat-in-time:</b></p> <p>None required, but material should be mixed for at least 5 minutes before use</p> <p><b>Shelf Life:</b></p> <p>Part A - 12 months, unopened Part B - 24 months, unopened Part F - 24 months, unopened Store indoors at 40°F to 100°F</p> <p><b>Flash Point (mixed):</b></p> <p>55°F</p> <p><b>Reducer/Clean up:</b></p> <p>Above 70°F: R2KT4, 150 Flash Naphtha Below 70°F: R2K4, Xylene</p>		@ 40°F	@ 77°F	@ 100°F	To touch:	25 minutes	20 minutes	5 minutes	To handle:	1 hour	20 minutes	15 minutes	To topcoat:	7 days	24 hours	8 hours	To cure:	7 days	36 hours	24 hours	To stack	6 hours	2 hours	1 hour	<p><b>System Tested:</b> (unless otherwise indicated) Substrate: Steel Surface Preparation: SSPC-SP10 1 ct. Zinc Clad II Plus @ 3.0 mils dft</p> <p><b>Adhesion:</b> Method: ASTM D4541 Result: 689 psi</p> <p><b>Direct Impact Resistance:</b> Method: ASTM D2794-92 Result: 60 in lbs.</p> <p><b>Dry Heat Resistance:</b> Method: ASTM D2485 Result: 750°F*</p> <p><b>Flexibility:</b> Method: ASTM D522, 180° bend, 1" mandrel Result: Passes</p> <p><b>Pencil Hardness:</b> Method: ASTM D3363 Result: 3H</p> <p><b>Salt Fog Resistance:</b> Method: ASTM B117, 7000 hours Result: Rating 9 per ASTM D714 for blistering Rating 9 per ASTM D610 for rusting</p> <p><b>Slip Coefficient (zinc only):</b> Method: AISC Specification for Structural Joints Using ASTM A325 or ASTM A490 Bolts Result: Class B, 0.67</p> <p><b>Slip Coefficient (system listed below):</b> 1 ct. Zinc Clad II Plus @ 2.0 - 4.0 mils dft 1 ct. Steel Spec Epoxy Primer @ 4.0 - 6.0 mils dft Method: AISC Specification for Structural Joints using ASTM A325 or ASTM A490 Bolts Result: Passes Class B, .56</p> <p>Provides performance comparable to products formulated to specifications Mil-P-38336 and Mil-P-46105.</p> <p>*Acceptable for use up to 1000°F when topcoated with Kern Hi-Temp Heat-Flex II 800 Aluminum.</p>	
	@ 40°F	@ 77°F	@ 100°F																							
To touch:	25 minutes	20 minutes	5 minutes																							
To handle:	1 hour	20 minutes	15 minutes																							
To topcoat:	7 days	24 hours	8 hours																							
To cure:	7 days	36 hours	24 hours																							
To stack	6 hours	2 hours	1 hour																							

Zinc Rich

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continued on back

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REPORT HI-2114830

Rev. 4

8.A-8

HI-STORM FW SYSTEM FSAR - Non-Proprietary  
Revision 4, June 24, 2015

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**Industrial  
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Coatings**

## 6.13

# ZINC CLAD® II PLUS

## INORGANIC ZINC-RICH COATING

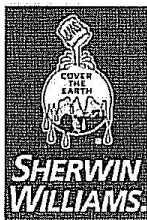
PART A B69VZ12  
PART B B69VZ13  
PART B B69VZ15  
PART F B69D11

BASE  
ACCELERATOR  
ACCELERATOR  
ZINC DUST

### PRODUCT INFORMATION

RECOMMENDED SYSTEMS	SURFACE PREPARATION
<b>Steel, Immersion:</b> 1 ct. Zinc Clad II Plus @ 2.0 - 4.0 mils dft  <b>Steel, Epoxy Topcoat, Atmospheric:</b> 1 ct. Zinc Clad II Plus @ 2.0 - 4.0 mils dft 1 ct. Macropoxy 646 @ 5.0 - 10.0 mils dft  <b>Steel, Polyurethane Topcoat, Atmospheric:</b> 1 ct. Zinc Clad II Plus @ 2.0 - 4.0 mils dft 1 ct. Macropoxy 646 @ 5.0 - 10.0 mils dft 1 ct. Acrolon 218 HS @ 3.0 - 6.0 mils dft  <b>Steel, Polyurethane Topcoat, Atmospheric:</b> 1 ct. Zinc Clad II Plus @ 2.0 - 4.0 mils dft 1 ct. Macropoxy 646 @ 5.0 - 10.0 mils dft 1 ct. Hi-Solids Polyurethane @ 3.0 - 4.0 mils dft  NOTE: 1 ct. of DTM Wash Primer can be used as an intermediate coat under recommended topcoats to prevent pinholing.  <b>Steel (Class B Compliant System):</b> 1 ct. Zinc Clad II Plus @ 2.0 - 4.0 mils dft 1 ct. Steel Spec Epoxy Primer, red @ 4.0 - 6.0 mils dft	Surface must be clean, dry, and in sound condition. Remove all oil, dust, grease, dirt, loose rust, and other foreign material to ensure adequate adhesion.  Refer to product Application Bulletin for detailed surface preparation information.  Minimum recommended surface preparation: Iron & Steel: Atmospheric: SSPC-SP6/NACE 3, 2 mil profile Immersion: SSPC-SP10/NACE 2, 2 mil profile
	TINTING
	Do not tint.
	APPLICATION CONDITIONS
	Temperature: 20°F minimum, 100°F maximum (air, surface, and material) At least 5°F above dew point  Relative humidity: 40% - 90% maximum Water misting may be required at humidities below 50%  Refer to product Application Bulletin for detailed application information.
	ORDERING INFORMATION
	Packaging: 3.66 gallons total, mixed Part A: 2.21 gallon kit Part B: 0.20 gallon Part F: 73 lbs zinc dust  Weight per gallon: 26.83 ± 0.2 lb, mixed
	SAFETY PRECAUTIONS
	Refer to the MSDS sheet before use.  Published technical data and instructions are subject to change without notice. Contact your Sherwin-Williams representative for additional technical data and instructions.
DISCLAIMER	WARRANTY
The information and recommendations set forth in this Product Data Sheet are based upon tests conducted by or on behalf of The Sherwin-Williams Company. Such information and recommendations set forth herein are subject to change and pertain to the product offered at the time of publication. Consult your Sherwin-Williams representative to obtain the most recent Product Data Information and Application Bulletin.	The Sherwin-Williams Company warrants our products to be free of manufacturing defects in accord with applicable Sherwin-Williams quality control procedures. Liability for products proven defective, if any, is limited to replacement of the defective product or the refund of the purchase price paid for the defective product as determined by Sherwin-Williams. NO OTHER WARRANTY OR GUARANTEE OF ANY KIND IS MADE BY SHERWIN-WILLIAMS, EXPRESSED OR IMPLIED, STATUTORY, BY OPERATION OF LAW OR OTHERWISE, INCLUDING MERCHANTABILITY AND FITNESS FOR A PARTICULAR PURPOSE.





**Industrial  
&  
Marine  
Coatings**

# 6.13A ZINC CLAD® II PLUS INORGANIC ZINC-RICH COATING

PART A	B69VZ12	BASE
PART B	B69VZ13	ACCELERATOR
PART B	B69VZ15	ACCELERATOR
PART F	B69D11	ZINC DUST

APPLICATION BULLETIN		Revised 12/05
SURFACE PREPARATION	APPLICATION CONDITIONS	
<p>Zinc rich coatings require direct contact between the zinc pigment in the coating and the metal substrate for optimum performance. Surface must be dry, free from oil, dirt, dust, mill scale or other contaminants to ensure adequate adhesion.</p> <p><b>Iron &amp; Steel (atmospheric service):</b> Remove all oil and grease from surface by Solvent Cleaning per SSPC-SP1. Minimum surface preparation is Commercial Blast Cleaning per SSPC-SP6/NACE 3. For better performance, use Near White Metal Blast Cleaning per SSPC-SP10/NACE 2. Blast clean all surfaces using a sharp, angular abrasive for optimum surface profile (2 mils). Prime any bare steel the same day as it is cleaned or before flash rusting occurs.</p> <p><b>Iron &amp; Steel (immersion service):</b> Remove all oil and grease from surface by Solvent Cleaning per SSPC-SP1. Minimum surface preparation is Near White Metal Blast Cleaning per SSPC-SP10/NACE 2. Blast clean all surfaces using a sharp, angular abrasive for optimum surface profile (2 mils). Remove all weld spatter and round all sharp edges by grinding. Prime any bare steel the same day as it is cleaned or before flash rusting occurs.</p> <p><b>Note:</b> If blast cleaning with steel media is used, an appropriate amount of steel grit blast media may be incorporated into the work mix to render a dense, angular 1.5 - 2.0 mil surface profile. This method may result in improved adhesion and performance.</p>	<p>Temperature: 20°F minimum, 100°F maximum (air, surface, and material) At least 5°F above dew point</p> <p>Relative humidity: 40% - 90% maximum Water misting may be required at humidities below 50%</p>	
APPLICATION EQUIPMENT		
<p>The following is a guide. Changes in pressures and tip sizes may be needed for proper spray characteristics. Always purge spray equipment before use with listed reducer. Any reduction must be compliant with existing VOC regulations and compatible with the existing environmental and application conditions.</p> <p><b>Reducer/Clean up</b> Above 70°F ..... R2KT4, 150 Flash Naphtha Below 70°F ..... R2K4, Xylene</p> <p><b>Airless Spray</b> (use Teflon packings and continuous agitation) Unit ..... Graco 30:1 Pressure ..... 2700 psi Hose ..... 3/8" ID Tip ..... .019" - .021" Filter ..... 30 mesh Reduction ..... As needed up to 4% by volume For continuous operation in larger areas, use Speedflo Airless Commander Zinc Pump. Set ball checks to maximum travel for viscous material.</p> <p><b>Conventional Spray</b> (continuous agitation required) Gun ..... Binks 95 Fluid Nozzle ..... 66 Fluid Hose ..... 1/2" ID, 50 ft maximum Air Nozzle ..... 63PB Air Hose ..... 1/2" ID, 50 ft maximum Atomization Pressure ... 25 psi Fluid Pressure ..... 10-20 psi Reduction ..... As needed up to 4% by volume</p> <p>Keep pressure pot at level of applicator to avoid blocking of fluid line due to weight of material. Blow back coating in fluid line at intermittent shutdowns, but continue agitation at pressure pot.</p> <p><b>Brush</b> ..... For touch up in small areas only</p> <p>If specific application equipment is not listed above, equivalent equipment may be substituted.</p>		

Zinc Rich

6.13A

continued on back

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**Industrial  
&  
Marine  
Coatings**

# 6.13A ZINC CLAD® II PLUS INORGANIC ZINC-RICH COATING

PART A B69VZ12  
PART B B69VZ13  
PART B B69VZ15  
PART F B69D11

BASE  
ACCELERATOR  
ACCELERATOR  
ZINC DUST

## APPLICATION BULLETIN

APPLICATION PROCEDURES	PERFORMANCE TIPS																								
<p>Surface preparation must be completed as indicated. Zinc Clad II Plus comes in premeasured containers, which when mixed provides ready-to-apply material.</p> <p><b>Mixing Instructions:</b> Thoroughly agitate Binder, Part A. Using continuous air driven agitation, slowly mix all of Zinc Dust, Part F, into all of Binder Part A until mixture is completely uniform. Continue agitation and add Part B. After mixing, pour mixture through 30-mesh screen. Mixed material must be used within 8 hours. Do not mix previously mixed material with new. No "sweat-in" period is required.</p> <p>If reducer solvent is used, add only after components have been thoroughly mixed.</p> <p>Continuous agitation of mixture during application is required, otherwise zinc dust will quickly settle out.</p> <p>Apply paint at the recommended film thickness and spreading rate as indicated below:</p> <p><b>Recommended Spreading Rate per coat:</b> Wet mils: 3.0 - 6.0 Dry mils: 2.0 - 4.0 Coverage: 400 - 610 sq ft/gal approximate</p> <p><b>Note:</b> Brush application is for small areas only. Application of coating above maximum or below minimum recommended spreading rate may adversely affect coating performance.</p> <p><b>Drying Schedule @ 4.0 mils wet @ 50% RH:</b></p> <table><tr><th></th><th>@40°F</th><th>@ 77°F</th><th>@100°F</th></tr><tr><td>To touch:</td><td>25 minutes</td><td>20 minutes</td><td>5 minutes</td></tr><tr><td>To handle:</td><td>1 hour</td><td>20 minutes</td><td>15 minutes</td></tr><tr><td>To topcoat:</td><td>7 days</td><td>24 hours</td><td>8 hours</td></tr><tr><td>To cure:</td><td>7 days</td><td>36 hours</td><td>24 hours</td></tr><tr><td>To stack</td><td>6 hours</td><td>2 hours</td><td>1 hour</td></tr></table> <p>Drying time is temperature, humidity, and film thickness dependent.</p> <p><b>Pot Life:</b> 8 hours @ 77°F High humidity will shorten pot life</p> <p><b>Sweat-in-time:</b> None required, but material should be mixed for at least 5 minutes before use</p>		@40°F	@ 77°F	@100°F	To touch:	25 minutes	20 minutes	5 minutes	To handle:	1 hour	20 minutes	15 minutes	To topcoat:	7 days	24 hours	8 hours	To cure:	7 days	36 hours	24 hours	To stack	6 hours	2 hours	1 hour	<p><b>Topcoating:</b> Note minimum cure times at normal conditions before topcoating. Longer drying periods are required if primer cannot be water mist sprayed when humidity is low. Water misting may be required at humidities below 50% to enhance cure rate.</p> <p>Occasionally topcoats will pinhole or delaminate from zinc-rich coatings. This is usually due to poor ambient conditions or faulty application of topcoats. This can be minimized by:</p> <ul style="list-style-type: none"><li>• Provide adequate ventilation and suitable application and substrate temperature.</li><li>• If pinholing develops during topcoating, apply a mist coat of the top coat, reduced up to 50%. Allow 10 minutes flash off and follow with a full coat.</li></ul> <p>An intermediate coat is recommended to provide uniform appearance of the topcoat.</p> <p>Stripe coat all crevices, welds, and sharp angles to prevent early failure in these areas.</p> <p>When using spray application, use a 50% overlap with each pass of the gun to avoid holidays, bare areas, and pinholes. If necessary, cross spray at a right angle.</p> <p>Spreading rates are calculated on volume solids and do not include an application loss factor due to surface profile, roughness or porosity of the surface, skill and technique of the applicator, method of application, various surface irregularities, material lost during mixing, spillage, overthinning, climatic conditions, and excessive film build.</p> <p>Excessive reduction of material can affect film build, appearance, and performance.</p> <p>Do not mix previously catalyzed material with new.</p> <p>Do not apply the material beyond recommended pot life.</p> <p>In order to avoid blockage of spray equipment, clean equipment before use or before periods of extended downtime with Reducer R2KT4, 150 Flash Naphtha.</p> <p>Keep pressure pot at level of applicator to avoid blocking of fluid line due to weight of material. Blow back coating in fluid line at intermittent shutdowns, but continue agitation at pressure pot.</p> <p>Application above recommended film thickness may result in mud cracking and poor topcoat appearance.</p> <p>During the early stages of drying, the coating is sensitive to rain, dew, high humidity, and moisture condensation. If possible, plan painting schedules to avoid these influences during the first 16-24 hours of curing.</p> <p>Topcoats may be applied once 50 MEK double rubs are achieved. No zinc or only slight traces should be visible. Coin hardness test can also be used.</p> <p>Refer to Product Information sheet for additional performance characteristics and properties.</p>
	@40°F	@ 77°F	@100°F																						
To touch:	25 minutes	20 minutes	5 minutes																						
To handle:	1 hour	20 minutes	15 minutes																						
To topcoat:	7 days	24 hours	8 hours																						
To cure:	7 days	36 hours	24 hours																						
To stack	6 hours	2 hours	1 hour																						
CLEAN UP INSTRUCTIONS	SAFETY PRECAUTIONS																								
<p>Clean spills and spatters immediately with Reducer R2KT4, 150 Flash Naphtha or R2K4, Xylene. Clean hands and tools immediately after use with Reducer R2KT4, 150 Flash Naphtha or R2K4, Xylene. Follow manufacturer's safety recommendations when using any solvent.</p>	<p>Refer to the MSDS before use.</p> <p>Published technical data and instructions are subject to change without notice. Contact your Sherwin-Williams representative for additional technical data and instructions.</p>																								
DISCLAIMER	WARRANTY																								
<p>The information and recommendations set forth in this Product Data Sheet are based upon tests conducted by or on behalf of The Sherwin-Williams Company. Such information and recommendations set forth herein are subject to change and pertain to the product offered at the time of publication. Consult your Sherwin-Williams representative to obtain the most recent Product Data Information and Application Bulletin.</p>	<p>The Sherwin-Williams Company warrants our products to be free of manufacturing defects in accord with applicable Sherwin-Williams quality control procedures. Liability for products proven defective, if any, is limited to replacement of the defective product or the refund of the purchase price paid for the defective product as determined by Sherwin-Williams. NO OTHER WARRANTY OR GUARANTEE OF ANY KIND IS MADE BY SHERWIN-WILLIAMS, EXPRESSED OR IMPLIED, STATUTORY, BY OPERATION OF LAW OR OTHERWISE, INCLUDING MERCHANTABILITY AND FITNESS FOR A PARTICULAR PURPOSE.</p>																								

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## CHAPTER 9: OPERATING PROCEDURES

### 9.0 INTRODUCTION

This chapter contains the operating procedures required for the dry storage of spent nuclear fuel at an on-site HI-STORM FW ISFSI. The decay heat, initial enrichment, burnup and cooling time of the SNF must accord with the restrictions in the Technical Specification. The unloading procedure is also described in this chapter. This sequence of activities is collectively referred to as short-term operations in this safety analysis report (SAR).

The procedures provided in this chapter are prescriptive to the extent that they provide the basis and general guidance for plant personnel in preparing detailed, written, site-specific, loading, handling, storage, and unloading procedures. Users may add, modify the sequence of, perform in parallel, or delete steps as necessary provided that the intent of this guidance are met and the requirements of the Certificate of Compliance (CoC) are complied with *literally*. The information provided in this chapter complies with the provisions of NUREG-1536 [9.0.1].

The information presented in this chapter along with the technical basis of the system design described in this SAR will be used to develop detailed operating procedures. Equipment specific operating details such as valve manipulation, canister drying method, special rigging, etc., will be provided to individual users of the system based on the specific ancillary equipment selected and the configuration of the site. In preparing the site-specific procedures, the user must consult the conditions of the CoC, equipment-specific operating instructions, and the plant's working procedures as well as the information in this chapter to ensure that the short-term operations shall be carried out with utmost safety and ALARA.

The following generic criteria shall be used to determine whether the site-specific operating procedures developed pursuant to the guidance in this chapter are acceptable for use:

- All heavy load handling instructions are in keeping with the guidance in industry standards, and Holtec-provided instructions.
- The procedures are in conformance with this FSAR and the COC.
- The operational steps are ALARA.
- The procedures contain provisions for documenting successful execution of all safety significant steps for archival reference.
- Procedures contain provisions for classroom and hands-on training and for a Holtec-approved personnel qualification process to ensure that all operations personnel are adequately trained.
- The procedures are sufficiently detailed and articulated to enable craft labor to execute them in *literal compliance* with their content.

The operations described in this chapter assume that the fuel will be loaded into or unloaded from the MPC submerged in a spent fuel pool. With some modifications, the information presented herein can be used to develop site-specific procedures for loading or unloading fuel into the system within a hot cell or other remote handling facility.

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Users are required to develop or modify existing programs and procedures to account for the implementation of the HI-STORM FW system. Written procedures are required to be developed or modified to account for such items as handling and storage of systems, structures and components identified as *important-to-safety*, heavy load handling, specialized instrument calibration, special nuclear material accountability, fuel handling procedures, training, equipment, and process qualifications. Users shall implement controls to ensure that all critical set points do not exceed the design limit of lifting equipment and appurtenances.

Control of the operation shall be performed in accordance with the user's Quality Assurance (QA) program to ensure critical steps are not overlooked and that the cask has been confirmed to meet all requirements of the CoC before being released for on-site storage under Part 72.

Fuel assembly selection and verification shall be performed by the user in accordance with written, approved procedures that ensure that only SNF assemblies authorized in the CoC are loaded into the MPC. Fuel handling shall be performed in accordance with written site-specific procedures.

ALARA notes and warnings in this chapter are included to alert users to radiological issues. Actions identified with these notes and warnings are of an advisory nature and shall be implemented based on a site-specific determination by radiation protection personnel.

Section 9.1 provides a technical basis for loading and unloading procedures. Section 9.2 provides the guidance for loading the HI-STORM FW system. Section 9.3 provides the procedures for ISFSI operations and general guidance for performing maintenance and responding to abnormal events. Responses to abnormal events that may occur during normal loading operations are provided with the procedure steps. Section 9.4 provides the procedure for unloading the HI-STORM FW system.

## 9.1 TECHNICAL AND SAFETY BASIS FOR LOADING AND UNLOADING PROCEDURES

The procedures herein are developed for the loading, storing, and unloading of spent fuel in the HI-STORM FW system. The activities involved in loading of spent fuel in a canister system, if not carefully performed, may present physical risk to the operations staff. The design of the HI-STORM FW system, including these procedures, the ancillary equipment and the Technical Specifications, serve to minimize potential risks and mitigate consequences of potential events.

The primary objective of the information presented in this chapter is to identify and describe the sequence of significant operations and actions that are important to safety for cask loading, cask handling, storage operations, and cask unloading to adequately protect health and minimize danger to life or property, protect the fuel from significant damage or degradation, and provide for the safe performance of tasks and operations.

In the event of an extreme abnormal condition the appropriate procedural guidance to respond to the situation must be available and ready for implementation. As a minimum, the procedures shall address establishing emergency action levels, implementation of emergency action program, establishment of personnel exclusions zones, monitoring of radiological conditions, actions to mitigate or prevent the release of radioactive materials, and recovery planning and execution and reporting to the appropriate regulatory agencies, as required.

Table 9.1.1

OPERATIONAL CONSIDERATIONS	
POTENTIAL EVENTS	METHODS USED TO ADDRESS AN ADVERSE EVENT
Cask Drop During Handling Operations	Cask lifting and handling equipment is designed to ANSI N14.6, as required.
Cask Tip-Over Prior to welding of the MPC lid	The design of the Lift Yoke prevents inadvertent disconnection during periods where it is attached.
Contamination of the MPC external shell	The annulus seal, bottom lid, and Annulus Overpressure System minimize the potential for the MPC external shell to become contaminated from contact with the spent fuel pool water.
Contamination spread from cask process system exhausts	Processing systems are equipped with exhausts that can be directed to the plant's processing systems.
Damage to fuel assembly cladding from oxidation	Fuel assemblies are not directly exposed to air or oxygen during loading and unloading operations. Fuel will be blanketed with an inert gas when not immersed in water. Water is introduced at a slow rate to avoid thermal shocking of the system.
Damage to Vacuum Drying System vacuum gauges from positive pressure	Vacuum gauges will be isolated from pressurized gas and water systems when not used for vacuum. Isolation valves allow gauges to be easily replaced in service.
Ignition of combustible mixtures of gas (e.g., hydrogen) during MPC lid welding or cutting	The area around MPC lid shall be appropriately monitored for combustible gases prior to and during welding or cutting activities. The space below the MPC lid shall be purged prior to and during these activities.
Excess dose from failed fuel assemblies during unloading operations	MPC gas sampling allows operators to determine the integrity of the fuel cladding prior to opening the MPC. This allows preparation and planning for failed fuel. 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 operation.
Excess dose to operators	The procedures provide ALARA Notes and Warnings when radiological conditions may change.

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Table 9.1.1	
OPERATIONAL CONSIDERATIONS	
POTENTIAL EVENTS	METHODS USED TO ADDRESS AN ADVERSE EVENT
Excess generation of radioactive waste	The HI-STORM FW system uses process systems 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. Where possible, items are installed by hand and require no tools.
Fuel assembly misloading event	Procedural guidance is given to perform assembly selection verification and a post-loading visual verification of assembly identification prior to installation of the MPC lid.
Incomplete moisture removal from MPC	The vacuum drying process reduces the MPC pressure in a controlled manner to prevent the formation of ice. Vacuum is held below 3 torr for 30 minutes with the vacuum pump isolated to assure dryness. If the forced helium dehydration process is used, the temperature of the gas exiting the demister is held below 21 °F for a minimum of 30 minutes. The TS require the surveillance requirement for moisture removal to be met before entering transport operations.
Incorrect MPC lid installation	Procedural guidance is given to visually verify correct MPC lid installation prior to HI-TRAC removal from the spent fuel pool.
Load Drop	Rigging diagrams and procedural guidance are provided to users for all applicable lifts. Component weights are provided to users on a site-specific basis. Heavy loads are handled in accordance with the guidance of NUREG-0612.
Over-pressurization of MPC during loading and unloading	Pressure relief devices in the water and gas processing systems limit the MPC pressure to acceptable levels.
Overstressing MPC lift lugs from side loading	Procedural guidance is provided for all heavy load handling activities on a site-specific basis.
Overweight cask lift	Procedural guidance is given to alert operators to potential overweight lifts. Site-specific weight evaluations are provided.
Personnel contamination by cutting/grinding activities	Procedural guidance is given to warn operators prior to cutting or grinding activities.

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Table 9.1.1	
OPERATIONAL CONSIDERATIONS	
POTENTIAL EVENTS	METHODS USED TO ADDRESS AN ADVERSE EVENT
Transfer cask carrying hot particles out of the spent fuel pool	Procedural guidance is given to scan the transfer cask prior to removal from the spent fuel pool.
Unplanned or uncontrolled release of radioactive materials	The MPC vent and drain ports are equipped with metal-to-metal seals to minimize the leakage during moisture removal and helium backfill operations. Unlike elastomer seals, the metal seals resist degradation due to temperature and radiation and allow future access to the MPC ports without hot tapping. The RVOAs allow the port to be opened and closed like a valve so gas sampling may be performed.

## 9.2 PROCEDURE FOR LOADING THE HI-STORM FW SYSTEM IN THE SPENT FUEL POOL

### 9.2.1 Overview of Loading Operations

The HI-STORM FW system is used to load, transfer, and store spent fuel. Specific steps, required to prepare the HI-STORM FW system for fuel loading, to load the fuel, to prepare the system for storage, and to place it in storage at an ISFSI are described in this chapter. The MPC transfer may be performed in the cask receiving area, at the ISFSI, or any other location deemed appropriate by the user. HI-TRAC VW and/or HI-STORM FW may be moved between the ISFSI and the fuel loading facility using any load handling equipment designed for such applications. Users of the HI-STORM FW system are required to develop detailed written procedures to control on-site transport operations. Instructions for general lifting, handling, and placement of the HI-STORM FW overpack, MPC, and HI-TRAC VW vary by site and are provided on a site-specific basis in Holtec-approved procedures and instructions.

The broad operational steps are explained below and illustrative figures are provided at the end of this section. At the start of loading operations, an empty MPC is upended. The empty MPC is raised and inserted into the HI-TRAC VW. The annulus is filled with plant demineralized water<sup>1</sup> and an inflatable seal is installed in the upper end of the annulus between the MPC and HI-TRAC VW to prevent spent fuel pool water from contaminating the exterior surface of the MPC when it is submerged in the pool. The MPC is filled with either spent fuel pool water or plant demineralized water (borated as required)<sup>2</sup>. The HI-TRAC VW top flange is outfitted with the lift blocks and the HI-TRAC VW and MPC are then raised and lowered into the spent fuel pool<sup>3</sup> for fuel loading using the lift yoke. Pre-selected assemblies<sup>4</sup> are loaded into the MPC and a visual verification of the assembly identification is performed.

While still underwater, a thick shielded lid (the MPC lid) is installed. The lift yoke remotely engages to the HI-TRAC VW lift blocks to lift the HI-TRAC VW and loaded MPC close to the spent fuel pool surface. When radiation dose rate measurements confirm that it is safe to remove the HI-TRAC VW from the spent fuel pool, the cask is removed from the spent fuel pool. The lift yoke and HI-TRAC VW are decontaminated, in accordance with instructions from the site's radiological protection personnel, as they are removed from the spent fuel pool.

HI-TRAC VW is placed in the designated preparation area and the lift yoke is removed. The next phase of decontamination is then performed. The top surfaces of the MPC lid and the upper flange of HI-TRAC VW are decontaminated. The neutron shield water jacket is filled with water (if drained). The inflatable annulus seal is removed and an annulus shield is installed. Dose rates are measured at the MPC lid to ensure that the dose rates are within expected values.

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<sup>1</sup> Users may substitute domestic water or radiologically clean borated water in each step where demineralized water is specified.

<sup>2</sup> Users may also fill the MPC with water during HI-TRAC placement in the spent fuel pool.

<sup>3</sup> Spent Fuel Pool as used in this chapter generically refers to the users designated cask loading location.

<sup>4</sup> Damaged fuel assemblies are loaded and stored in Damaged Fuel Containers in the MPC basket.

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The MPC water level and annulus water level are lowered slightly, the MPC is vented, and the MPC lid is welded on using the automated welding system. Visual examinations are performed on the tack welds. Liquid penetrant (PT) examinations are performed on the root and final passes. A progressive PT examination as described in the Code Alternatives listed in the CoC 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 the 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 measurement or determined analytically. The remaining bulk water in the MPC is drained.

Depending on the burn-up or decay heat load of the fuel to be loaded in the MPC, moisture is removed from the MPC using either a vacuum drying system (VDS) or forced helium dehydration (FHD) system. For MPCs without high burn-up fuel or with high burnup fuel and with sufficiently low decay heat, the vacuum drying system may be connected to the MPC and used to remove all liquid water from the MPC. The annular gap between the MPC and HI-TRAC is filled with water during vacuum drying to promote heat transfer from the MPC and maintain lower fuel cladding temperatures. The internal pressure is reduced and held in accordance with Technical Specifications to ensure that all liquid water is removed.

An FHD system is required for high-burn-up fuel at higher decay heat (it can be used as an alternative to vacuum drying) 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 demister is maintained in accordance with Technical Specification requirements to ensure that all liquid water is removed.

Following MPC moisture removal, by VDS or FHD, the MPC is backfilled with a predetermined amount of helium gas. The helium backfill ensures adequate heat transfer during storage, and provides an inert atmosphere for long-term fuel integrity. 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 (for multi-pass welds). The cover plate welds are then leak tested.

The MPC closure ring is then placed on the MPC and 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 and final welds are inspected using the liquid penetrant examination technique to ensure weld integrity.

The annulus shield (if utilized) is removed and the remaining water in the annulus is drained. The MPC lid and accessible areas of the top of the MPC shell are smeared for removable contamination. HI-TRAC VW surface dose rates are measured in accordance with the technical specifications. The MPC lift attachments are installed on the MPC lid. The MPC lift attachments

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are the primary lifting point on the MPC. MPC slings are installed between the MPC lift attachments and the lift yoke.

MPC transfer may be performed inside or outside the fuel building. The empty HI-STORM FW overpack is inspected and positioned with the lid removed. Next, the mating device is positioned on top of the HI-STORM FW and HI-TRAC VW is placed on top of it. The mating device assists in the removal of the HI-TRAC VW bottom lid and helps guide the HI-TRAC VW during its placement on the HI-STORM FW. The MPC slings are attached to the MPC lift attachments. The MPC is transferred using a suitable load handling device.

Next, the HI-TRAC VW bottom lid is removed and the mating device drawer is opened. The MPC is transferred into HI-STORM FW. Following verification that the MPC is fully lowered, the MPC slings are disconnected from the lifting device and lowered onto the MPC lid. Next, the HI-TRAC VW is removed from the top of HI-STORM FW<sup>5</sup>. The MPC slings and MPC lift attachments are removed. Plugs are installed in the empty MPC lifting holes to fill the voids left by the lift attachment bolts. Next, the mating device is removed. The HI-STORM FW lid, along with the temperature elements (if used), and vent screens may be installed at anytime after the mating device is removed. The HI-STORM FW is secured to the transporter (as applicable) and moved to the ISFSI pad. The HI-STORM FW overpack and HI-TRAC VW transfer cask may be moved using a number of methods as long as the lifting equipment requirements of this FSAR are met. Finally, the temperature elements connections are installed (if used), final dose rate measurements are taken, and any thermal testing (if required) is performed to ensure that the system is functioning within its design parameters.

### 9.2.2 Preparation of HI-TRAC VW and MPC

**Note:**

Handling of loaded equipment shall only be performed if the ambient temperature is above 0°F

1. Place HI-TRAC VW in the cask receiving area.
2. Perform a HI-TRAC VW receipt inspection and cleanliness inspection (See Table 9.2.5 for example).
3. Clear the HI-TRAC VW top for installation of the MPC.
4. Remove any road dirt. Remove any foreign objects from cavity locations.
5. If necessary, perform a radiological survey of the inside of HI-TRAC VW to verify there is no residual contamination from previous uses of the cask.
6. If necessary, configure HI-TRAC VW with the bottom lid.

<sup>5</sup> The empty HI-TRAC VW may be removed from the mating device with its bottom lid installed or removed.



7. Perform an MPC receipt inspection and cleanliness inspection (See Table 9.2.4 for example).
8. Install the MPC inside HI-TRAC VW in accordance with site-approved rigging procedures.
9. If necessary, perform an MPC, lid, closure ring, drain line, vent, and drain port cover plate fit test and verify that the weld prep is in accordance with the approved fabrication drawings.

**Note:**

Annulus filling and draining operations vary by site. Instructions for filling and draining the annulus along with the use of the annulus overpressure system are provided on a site-specific basis.

10. Fill the annulus with non-contaminated water to just below the inflatable seal seating surface.
11. Install the inflatable annulus seal around the MPC.
12. To the extent practicable, apply waterproof tape over any empty bolt holes or locations where water may create a decontamination issue.

**Note:**

Canister filling and draining operations vary by site. Instructions are provided on a site-specific basis.

13. Fill the MPC with water to approximately 12 inches below the top of the MPC shell. Refer to LCO 3.3.1 for boron concentration requirements.

**ALARA Note:**

Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

14. Place HI-TRAC VW in the designated cask loading area.
15. Verify spent fuel pool for boron concentration requirements in accordance with LCO 3.3.1. Testing must be completed within four hours prior to loading and every 48 hours after in accordance with the LCO. Two independent measurements shall be taken to ensure that the requirement of 10 CFR 72.124(a) is met.

### 9.2.3 MPC Fuel Loading

**Note:**

When loading an MPC requiring soluble boron, the boron concentration of the water shall be checked in accordance with LCO 3.3.1 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 the Approved Contents Section of Appendix B to the CoC, have been selected for loading into the MPC. Perform a verification of the types, amounts, and location of non-fuel hardware using plant fuel records to ensure that only non-fuel hardware that meet the conditions for loading, as specified in the Approved Contents Section of Appendix B to the CoC, 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<sup>6</sup>.
3. Perform a post-loading visual verification of the assembly identification to confirm that the serial numbers match the approved fuel loading pattern.
4. If required, install fuel shims where necessary in the fuel cells.

### 9.2.4 MPC Closure

1. Install MPC lid and remove the HI-TRAC VW from the spent fuel pool as follows:
  - a. Rig the MPC lid for installation in the MPC in accordance with site-approved rigging procedures.
  - b. Install the drain line to the underside of the MPC lid.
  - c. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC for installation.
  - d. Seat the MPC lid in the MPC and visually verify that the lid is properly installed.
  - e. Record the time to begin the time-to-boil monitoring, if necessary.
  - f. Engage the lift yoke to HI-TRAC VW.

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<sup>6</sup> Damaged fuel must be loaded into Damage Fuel Containers in the MPC basket.

**ALARA Note:**

Activated debris may have settled on the top face of HI-TRAC VW 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. Soluble boron concentration, when applicable, shall be monitored to prevent non-compliance with the Technical Specification LCO 3.3.1.

- g. Raise the HI-TRAC VW 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. Remove any activated or highly radioactive particles from the HI-TRAC VW or MPC.
- h. Continue to raise the HI-TRAC VW under the direction of the plant's radiological control personnel. Continue general decontamination activities.
- i. Remove HI-TRAC VW from the spent fuel pool while performing outer decontamination activities in accordance with directions from the radiological control personnel.
- j. Place HI-TRAC VW in the designated cask preparation area.

**Note:**

If the transfer cask is expected to be operated in an environment below 32 °F, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with clean potable or demineralized water. Depending on weight limitations, the neutron shield jacket may remain filled (with pure water or 25% ethylene glycol solution, as required). Cask weights shall be evaluated to ensure that the equipment load limitations are not violated.

- k. If previously drained, fill the neutron shield jacket with plant demineralized water or an ethylene glycol solution (25% ethylene glycol) as necessary.
- l. Disconnect any special rigging from the MPC lid and disengage the lift yoke in accordance with site-approved rigging procedures.

**Warning:**

MPC lid dose rates are measured to ensure that dose rates are within expected values. Dose rates exceeding the expected values could be an indication that fuel assemblies not meeting the CoC have been loaded.

- m. Measure the dose rates at the MPC lid and verify that the combined gamma and neutron dose is below expected values.
- n. Perform decontamination and a dose rate/contamination survey of HI-TRAC.
- o. Prepare the MPC annulus for MPC lid welding by removing the annulus seal and draining the annulus approximately 6 inches.

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2. Prepare for MPC lid welding as follows:

- a. Clean the vent and drain ports to remove any dirt or standing water. Install the RVOAs to the MPC lid vent and drain ports, leaving caps open.
- b. Lower the MPC internal water level in preparation for MPC lid-to-shell welding.

**ALARA Note:**

The MPC exterior shell survey is performed. Indications of contamination could require the MPC to be unloaded. In the event that the MPC shell is contaminated, users must decontaminate the annulus. If the contamination cannot be reduced to acceptable levels, the MPC must be returned to the spent fuel pool and unloaded. The MPC may then be removed and the external shell decontaminated.

- c. Survey the MPC lid top surfaces and the accessible areas (approximately the top three inches) of the MPC external shell. Decontaminate the MPC lid and accessible surfaces of the MPC shell in accordance with LCO 3.2.1.

3. Weld the MPC lid as follows:

- a. As necessary, install the MPC lid shims around the MPC lid to make the weld gap uniform and to close the gap to the requirements of the licensing drawings.
- b. Install the Automated Welding System (AWS).

**Note:**

It may be necessary to remove the RVOAs to allow access for the automated welding system. In this event, the vent and drain port caps should be opened to allow for thermal expansion of the MPC water.

**Caution:**

A radiolysis of water may occur in high flux conditions inside the MPC creating combustible gases. 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 purged with inert gas prior to, and during MPC lid welding operations, including welding, grinding, and other hot work, to provide additional assurance that flammable gas concentrations will not develop in this space.

- c. Perform combustible gas monitoring and purge the space under the MPC lid with an inert gas to ensure that there is no combustible mixture present in the welding area.

**Note:**

MPC closure welding procedures dictate the performance requirements and acceptance requirements of the weld examinations.

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- d. Perform the MPC lid-to-shell weld and NDE in accordance with the licensing drawings using approved procedures. Repair any weld defects in accordance with the applicable code and re-perform the NDE until the weld meets the required acceptance criteria.
4. Perform MPC lid-to-shell weld pressure testing in accordance with site-approved procedures.
5. Repeat the liquid penetrant examination on the final pass of the MPC lid-to-shell weld.
  - a. Repair any weld defects in accordance with the applicable code requirements and re-perform the NDE in accordance with approved procedures.
6. Drain the MPC and terminate time-to-boil monitoring and boron sampling program, where required.

**Note:**

Detailed procedures for MPC drying are provided on a site-specific basis. The following summarize those procedures.

7. Dry and backfill the MPC (Vacuum Drying Method).

**Note:**

During drying activities, the annulus between the MPC and the HI-TRAC VW must be maintained full of water. Water lost due to evaporation or boiling must be replaced to maintain the water level.

- a. Fill the annulus between the MPC and HI-TRAC VW with clean water. The water level must be within 6" of the top of the MPC.
- b. Attach the vacuum drying system (VDS) to the vent and drain port RVOAs. Other equipment configurations that achieve the same results may also be used.

**Caution:**

Rapidly reducing the pressure in the VDS piping and MPC while the system contains significant amounts of water can lead to freezing of the water and to improper conclusions that the system is dry. To prevent freezing of water, the MPC internal pressure should be lowered in a controlled fashion. The vacuum drying system pressure will remain at about 30 torr until most of the liquid water has been removed from the MPC.

- c. Start the VDS system and slowly reduce the MPC pressure to below 3 torr.

**Note:**

Helium backfill shall be in accordance with the Technical Specification using 99.995% (minimum) purity. If at any time during final closure operations the helium backfill gas is lost or oxidizing gases are introduced into the MPC, then the dryness test shall be repeated and the MPC refilled with helium in accordance with the Technical Specifications.

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- d. Perform the MPC drying pressure test in accordance with the Technical Specifications.
- e. When the MPC is dry, in accordance with the acceptance criteria in the LCO 3.1.1, close the vent and drain port valves.
- f. Backfill the MPC in accordance with LCO 3.1.1 using site-specific procedures.
- g. Disconnect the VDS from the MPC.
- h. Close the drain port RVOA cap and remove the drain port RVOA.
- i. If used, stop the water flow through the annulus between the MPC and HI-TRAC. Drain.
- j. Close the vent port RVOA and disconnect the vent port RVOA.

8. Dry and Backfill the MPC (FHD Method):

<b>Note:</b>
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 high purity helium.
<b>Note:</b>
MPC internal pressure during FHD operation must comply with Technical Specification.
<b>Caution:</b>
MPC internal pressure during FHD operation may be less than the Technical Specification minimum backfill requirement. In the event of an FHD System failure where the MPC internal pressure is below the Technical Specification limit, the MPC internal pressure must be raised to at least 20 psig 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. Drain the water from the annulus.
- c. Circulate the drying gas through the MPC while monitoring the circulating gas for moisture. Collect and remove the moisture from the system as necessary.
- d. Continue the monitoring and moisture removal until LCO 3.1.1 is met for MPC dryness.

<b>Note:</b>
The demister module must maintain the temperature of the helium exiting the FHD below the Technical Specification limits continuously from the end of the drying operations until the MPC has been backfilled and isolated. If the temperature of the gas exiting the FHD exceeds the temperature limit, the dryness test must be repeated and the backfill re-performed.

- e. Continue operation of the FHD system with the demohisturizer on.
  - f. 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 LCO 3.1.1.
  - g. Open the FHD bypass line and 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.
9. Weld the vent and drain port cover plates and perform NDE in accordance with the licensing drawings using approved procedures. Repair any weld defects in accordance with the applicable code and re-perform the NDE until the weld meets the required acceptance criteria.
10. Perform a leakage test of the MPC vent port cover plate and drain port cover plate in accordance with the following and site-approved procedures:
- a. If necessary, remove the cover plate set screws.
  - b. Flush the cavity with helium to remove the air and immediately install the set screws recessed approximately  $\frac{1}{4}$  inch below the top of the cover plate.
  - c. Plug weld the recess above each set screw to complete the penetration closure welding in accordance with the licensing drawings using approved procedures. Repair any weld defects in accordance with the applicable code and re-perform the NDE until the weld meets the required acceptance criteria.
  - d. Flush the area around the vent and drain cover plates with compressed air or nitrogen to remove any residual helium gas.
  - e. 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 leakage test methods and procedures of ANSI N14.5 [9.1.2]. The MPC Helium Leak Rate acceptance criterion is provided in LCO 3.1.1.
11. Weld the MPC closure ring as follows:
- a. Install and align the closure ring.
  - b. Weld the closure ring to the MPC shell and the MPC lid, and perform NDE in accordance with the licensing drawings using approved procedures. Repair any weld defects in accordance with the applicable code and re-perform the NDE until the weld meets the required acceptance criteria.
  - c. If necessary, remove the AWS.

### 9.2.5 Preparation for Storage

**ALARA Warning:**

Dose rates will rise around the top of the annulus as water is drained from the annulus. Apply appropriate ALARA practices.

**Caution:**

Limitations for the handling an MPC containing high burn-up fuel in a HI-TRAC VW are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to SAR Chapter 4.

1. Drain the remaining water from the annulus.
2. Perform the HI-TRAC VW surface dose rate measurements in accordance with the Technical Specifications. Measured dose rates must be compared with calculated dose rates that are consistent with the calculated doses that demonstrate compliance with the dose limits of 10CFR 72.104(a). Remove any surface contamination from the HI-TRAC surfaces as required by LCO 3.2.1.

**Note:**

HI-STORM FW receipt inspection and preparation may be performed independent of procedural sequence, but prior to transfer of the loaded MPC. See Table 9.2.3 for example of HI-STORM FW Receipt Inspection Checklist.

3. Perform a HI-STORM FW receipt inspection and cleanliness inspection in accordance with a site-approved inspection site-approved inspection checklist, if required.

**Note:**

MPC transfer may be performed at any location deemed appropriate by the licensee. The following steps describe the general transfer operations. The HI-STORM FW may be positioned on an air pad, roller skid or any other suitable equipment in the cask receiving area or at the ISFSI. The HI-STORM FW or HI-TRAC VW may be transferred to the ISFSI using any equipment specifically designed for such a function. The licensee is responsible for assessing and controlling floor loading conditions during the MPC transfer operations. Installation of the lid, vent screen, and other components may vary according to the cask movement methods and location of MPC transfer.

### 9.2.6 Placement of HI-STORM FW into Storage

1. Position an empty HI-STORM FW module at the designated MPC transfer location.
2. Remove any road dirt with water. Remove any foreign objects from cavity locations.
3. Transfer the HI-TRAC VW to the MPC transfer location.



4. Install the mating device on top of the HI-STORM FW.
5. Position HI-TRAC VW above HI-STORM FW.
6. Align HI-TRAC VW over HI-STORM FW and mate the components.
7. Attach the MPC to the lifting device in accordance with the site-approved rigging procedures.
8. Raise the MPC slightly to remove the weight of the MPC from the mating device.
9. Remove the bottom lid from HI-TRAC VW using the mating device.

**ALARA Warning:**

Personnel should remain clear (to the maximum extent practicable) of the HI-STORM FW annulus when HI-TRAC VW is removed due to radiation streaming. The mating device may be used to supplement shielding during removal of the MPC lift rigging.

10. Lower the MPC into HI-STORM FW.
11. Disconnect the MPC lifting slings from the lifting device.

**Note:**

It may be necessary, due to site-specific circumstances, to move HI-STORM FW from under the empty HI-TRAC VW to install the HI-STORM FW lid, while inside the Part 50 facility. In these cases, users shall evaluate the specifics of their movements within the requirements of their Part 50 license.

12. Remove HI-TRAC VW from on top of HI-STORM FW with or without the HI-TRAC bottom lid.
13. Remove the MPC lift rigging and install plugs in the empty MPC bolt holes.
14. Place HI-STORM FW in storage as follows:

**Note:**

Closing the mating device drawer while the MPC is in the HI-STORM will block air flow. The mating device drawer shall remain open, to the extent possible, such that the open air path is at least as large as the HI-STORM Lid vent openings until the mating device is to be removed from the HI-STORM. When the mating device drawer is closed for mating device removal, the process shall be completed in an expeditious manner.

- a. Remove the mating device.
- b. Inspect the HI-STORM FW lid studs and nuts or lid closure bolts for general condition. Replace worn or damaged components with new ones.

**Note:**

Unless the lift has redundant drop protection features (or equivalent safety factor) for the HI-STORM FW lid, the lid shall be kept less than 2 feet above the top surface of the overpack. This is performed to protect the MPC lid from a potential HI-STORM FW lid drop.

- c. Install the HI-STORM FW lid and the lid studs and nuts or lid closure bolts.
- d. Remove the HI-STORM FW lid lifting device and install the hole plugs in the empty holes. Store the lifting device in an approved plant storage location.

**Warning:**

HI-STORM FW dose rates are measured to ensure they are within expected values. Dose rates exceeding the expected values could indicate that fuel assemblies not meeting the CoC may have been loaded.

- e. Perform the HI-STORM FW surface dose rate measurements in accordance with the Technical Specifications. Measured dose rates must be compared with calculated dose rates that are consistent with the calculated doses that demonstrate compliance with the dose limits of 10CFR72.104(a).
- f. Secure HI-STORM FW to the transporter device as necessary.

**Note:**

The site-specific transport route conditions must satisfy the requirements of the Technical Specification.

- g. Perform a transport route walkdown to ensure that the transport conditions are met.
  - h. Transfer the HI-STORM FW to its designated storage location at the appropriate pitch.
  - i. Attach the HI-STORM FW temperature elements (if used) and screens.
15. If required per CoC Condition #8 the user must perform the following annular air flow thermal test or cite a test report that was performed and prepared by another user.
- a. The annular air flow thermal test shall be conducted at least 7 days after the HI-STORM is loaded in order for the overpack to establish thermal equilibrium.
  - b. The user or other qualified engineer shall calculate and record the actual heat load of the fuel stored in the HI-STORM.
  - c. To minimize the effects on the annular air flow, the test shall be performed when the weather is relatively dry and calm.
  - d. The ambient air temperature at the cask shall be recorded.

- e. The test data shall be collected for the annular flow between the MPC and HI-STORM inner shell as follows:
  - 1. The outlet vent screen shall be removed from one outlet vent, if necessary for instrument access. Alternatively, if access ports have been provided in the HI-STORM lid, the access port plugs may be removed and access ports used for instrument access.
  - 2. A hot wire anemometer or similar flow measuring instrument shall be inserted into the annular space between the MPC and HI-STORM inner shell.
  - 3. The flow measuring instrument shall be positioned at least 6" below the top of the MPC and shall not significantly block the air flow.
  - 4. The instrument shall not be placed too close to the MPC or HI-STORM shells to avoid edge effects on the flow.
  - 5. The outlet gamma shield and vent screen shall be re-installed if removed.
  - 6. Measurements of the air flow shall be taken and recorded for a minimum of three places radially across the annular gap.
  - 7. The outlet vent screen and gamma shield shall be removed from the outlet vent, if necessary, and the flow measuring instrument removed.
  - 8. The outlet gamma shield and vent screen shall be re-installed if removed.
  - 9. Re-install access port plugs if removed.
- f. Air flow in each of the three remaining outlet vents or access ports shall be measured and recorded in accordance with step 15.e above.
- g. All test data shall be transmitted to the general license holder for evaluation and validation of the thermal model.
- h. Users shall forward test and analysis results to the NRC in accordance with 10 CFR 72.4.

Table 9.2.1

**HI-STORM FW SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION**

<b>Equipment</b>	<b>Important To Safety Classification</b>	<b>Description</b>
Air Pads/Rollers	Not Important To Safety	Used for HI-STORM FW or HI-TRAC VW cask positioning. May be used in conjunction with the cask transporter or other HI-STORM FW or HI-TRAC VW lifting device.
Annulus Overpressure System	Not Important To Safety	The Annulus Overpressure System is used for protection against spent fuel pool water contamination of the external MPC shell and baseplate surfaces by providing a slight annulus overpressure during in-pool operations.
Automated Welding System	Not Important To Safety	Used for remote field welding of the MPC.
Cask Transporter	Not Important to Safety unless used for MPC transfers	Used for handling of the HI-STORM FW overpack and/or the HI-TRAC VW Transfer 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 a function. May also be used for MPC transfers if appropriately configured.
Lid and empty component lifting rigging	Not Important To Safety, Rigging shall be provided in accordance with NUREG 0612	Used for rigging components such as the HI-TRAC VW top lid, bottom lid, MPC lid, AWS, and HI-STORM FW Lid and the empty MPC.
Helium Backfill System	Not Important To Safety	Used for controlled insertion of helium into the MPC for pressure testing, blowdown and placement into storage.
HI-STORM FW Special Lifting Device	Determined site-specifically based on type, location, and height of lift being performed. Special lifting devices shall be provided in accordance with ANSI N14.6.	A special lifting device used for connecting the crane (or other primary lifting device) to the HI-STORM FW for cask handling.
HI-TRAC VW Lift Yoke/Lifting Links	Determined site-specifically based on type and location, and height of lift being performed. Lift yoke and lifting devices for loaded HI-TRAC VW handling shall be provided in accordance with ANSI N14.6.	Used for connecting the crane (or other primary lifting device) to the HI-TRAC VW for cask handling. Does not include the crane hook (or other primary lifting device). May include one or more extensions to prevent immersion of the crane hook into the spent fuel pool water.
HI-TRAC VW transfer frame	Not Important To Safety	A steel frame used to support HI-TRAC VW during delivery, on-site movement and upending/downending operations.
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.

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Table 9.2.1

HI-STORM FW SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION		
Equipment	Important To Safety Classification	Description
MPC Lift Attachments	Important To Safety – Category A. MPC Lift Attachments shall be provided in accordance with ANSI N14.6.	MPC lift attachments consist of the strongback and attachment hardware. The MPC lift attachments are used to support the MPC during MPC transfer from HI-TRAC VW into HI-STORM FW and vice versa. The ITS classification of the lifting device attached to the attachments may be lower than the attachment itself, as determined site-specifically. Lift Attachments may take different forms based on site specific needs and may include remote disconnect features.
Pressure Test System	Not Important to Safety	Used to pressure test the MPC lid-to-shell weld.
HI-TRAC Lift Block	Important-To-Safety Category A. Lift Blocks shall be provided in accordance with ANSI N14.6.	Used to attach the HI-TRAC to the lifting yoke.
Mating Device	Important-To-Safety – Category B	Used to mate HI-TRAC VW to HI-STORM FW during transfer operations. Used to shield operators during MPC transfer operations. Includes sliding drawer for use in removing HI-TRAC VW bottom lid.
MPC Lifting Slings	Important To Safety – Category A – Rigging shall be provided in accordance with NUREG 0612.	Used to secure the MPC to the overhead lifting device during HI-TRAC VW bottom lid removal and MPC transfer operations. Attaches between the MPC lift attachments and the lift yoke or overhead lifting device.
MPC Upending Device	Not Important to Safety	Used to evenly support the MPC during handling and upending operations and help control the upending process.
MSLD (Helium Leakage Detector)	Not Important to Safety	Used for helium leakage testing of the MPC closure welds.
Vacuum Drying System	Not Important To Safety	Used for removal of residual moisture from the MPC following water draining.
Forced Helium Dehydration System	Not Important To Safety	Used for removal of residual moisture from the MPC following water draining.
Vent and Drain RVOAs	Not Important To Safety	Used to access the vent and drain ports. The vent and drain RVOAs allow the vent and drain ports to be operated like valves.
Weld Removal System	Not Important To Safety	Semi-automated weld removal system used for removal of the MPC field weld to support unloading operations.

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Table 9.2.2

**HI-STORM FW SYSTEM INSTRUMENTATION SUMMARY FOR LOADING AND  
UNLOADING OPERATIONS<sup>†</sup>**

<b>Instrument</b>	<b>Function</b>
Contamination Survey Instruments	Monitors fixed and non-fixed contamination levels.
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 fluid flow rate during various loading and unloading operations.
Helium Mass Spectrometer Leakage Detector (MSLD)	Ensures leakage rates of welds are within acceptable limits.
Volumetric Examination Testing Rig	Used to assess the integrity of the MPC lid-to-shell weld.
Pressure Gauges	Ensures correct pressure during loading and unloading operations.
Temperature Gauges	Monitors the state of gas and water temperatures during closure and unloading operations.
Vacuum Gages (Optional)	Used for vacuum drying operations and to prepare an MPC evacuated sample bottle for MPC gas sampling for unloading operations.
Moisture Monitoring Instruments	Used to monitor the MPC moisture levels as part of the moisture removal system.

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<sup>†</sup> All instruments require calibration. See figures at the end of this section for additional instruments, controllers and piping diagrams.

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Table 9.2.3

## HI-STORM FW SYSTEM OVERPACK INSPECTION CHECKLIST

**Note:**

This checklist provides the basis for establishing a site-specific inspection checklist for the HI-STORM FW overpack. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

HI-STORM FW Overpack Lid:

1. Lid studs and nuts or lid closure bolts shall be inspected for general condition.
2. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
3. All lid surfaces shall be relatively free of dents, scratches, gouges or other damage.
4. The lid shall be inspected for the presence or availability of studs and nuts and hole plugs.
5. Lid lifting device holes shall be inspected for dirt and debris and thread condition.
6. Lid bolt holes shall be inspected for general condition.
7. Vent screens shall be inspected for proper fit and for tears and holes that would allow debris entry into the vent openings.
8. Vent openings shall be inspected for foreign material that may cause vent blockage.

HI-STORM FW Main Body:

1. Lid bolt holes shall be inspected for dirt, debris, and thread condition.
2. Vents shall be free from obstructions.
3. Vent screens shall be inspected for proper fit and for tears and holes that would allow debris entry into the vent openings.
4. The interior cavity shall be free of debris, litter, tools, and equipment.
5. Painted surfaces shall be inspected for corrosion, and chipped, cracked or blistered paint.
6. The nameplate shall be inspected for presence, legibility, and general condition and conformance to Quality Assurance records package.

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Table 9.2.4

## MPC INSPECTION CHECKLIST

**Note:**

This checklist provides the basis for establishing a site-specific inspection checklist for MPC. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

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. Vent and Drain attachments shall be inspected for availability, thread condition operability, and general condition.
4. 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.
7. The MPC lid lift holes shall be inspected for thread condition.
8. The MPC lid, cover plates, and closure ring shall be checked for proper fit-up.

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. Basket panels shall be inspected for gross deformation that may inhibit fuel assembly insertion.
4. Lift lugs shall be inspected for general condition.
5. Lift lug threads shall be inspected for thread condition
6. Verify proper MPC basket type for contents.
7. Serial numbers shall be inspected for readability.



Table 9.2.5

## HI-TRAC VW TRANSFER CASK INSPECTION CHECKLIST

**Note:**

This checklist provides the basis for establishing a site-specific inspection checklist for the HI-TRAC VW Transfer Cask. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation, and potential corrective action prior to use.

HI-TRAC VW Main Body:

1. The painted surfaces shall be inspected for corrosion, chipped, cracked, or blistered paint.
2. 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.
3. The nameplate shall be inspected for presence and general condition.
4. The neutron shield jacket shall be inspected for leaks.
5. Neutron shield jacket pressure relief device shall be inspected for presence and general condition.
6. The neutron shield jacket fill and neutron shield jacket drain plugs shall be inspected for presence, leaks, and general condition.
7. Bottom lid flange surface shall be clean and free of large scratches and gouges that may inhibit sealing of the lid to body.
8. The threaded anchor locations shall be inspected for thread damage, excessive wear, and general condition.

HI-TRAC VW Bottom lid:

1. Seal shall be inspected for cracks, breaks, cuts, excessive wear, flattening, and general condition.
2. Drain line shall be inspected for blockage and thread condition.
3. The lifting holes shall be inspected for thread damage.
4. The bolts shall be inspected for indications of overstressing (i.e., cracks and deformation, thread damage, and excessive wear).
5. The painted surfaces shall be inspected for corrosion, chipped, cracked, or blistered paint.
6. Threads shall be inspected for indications of damage.

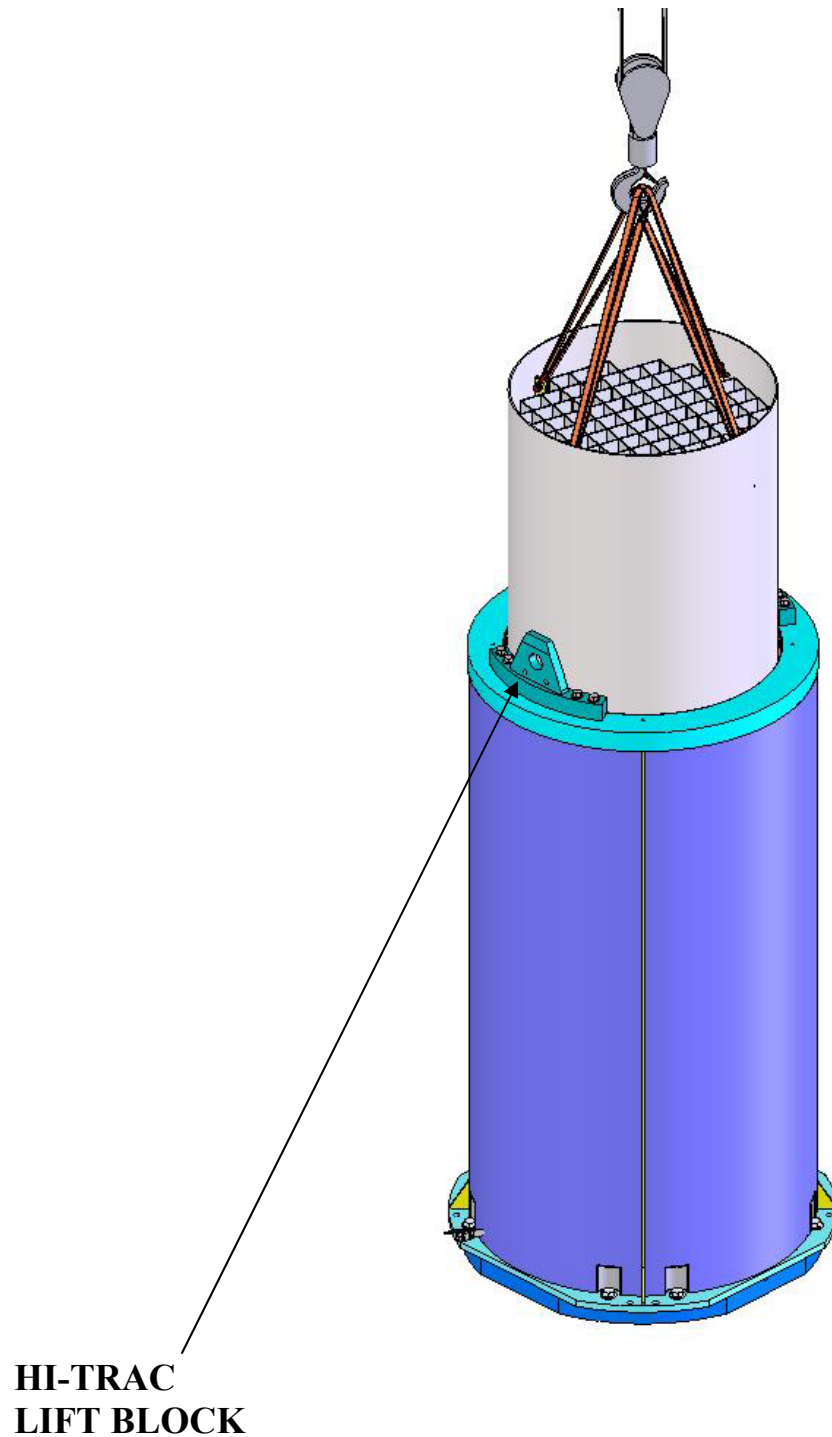
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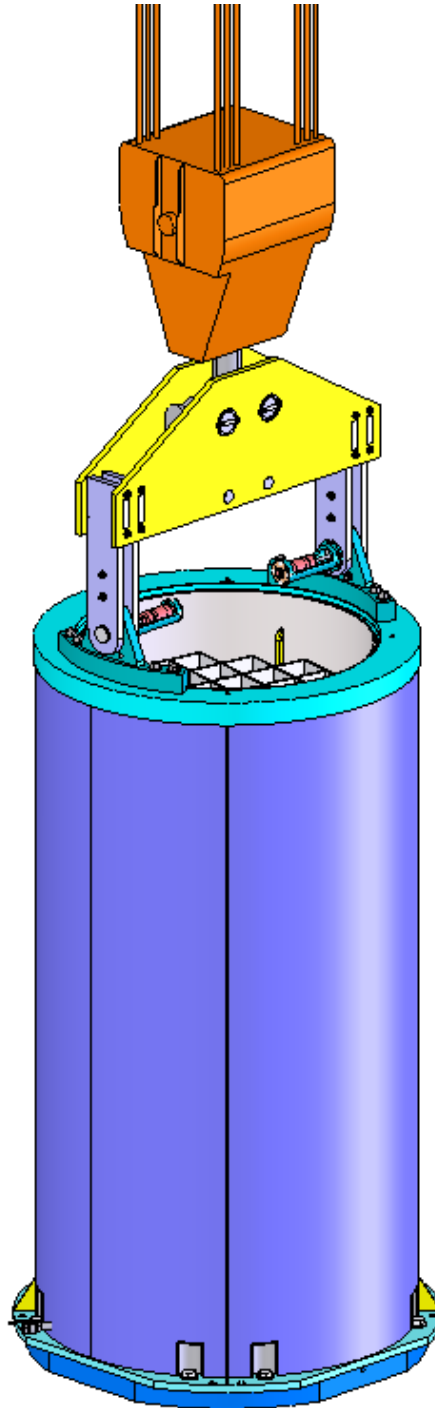
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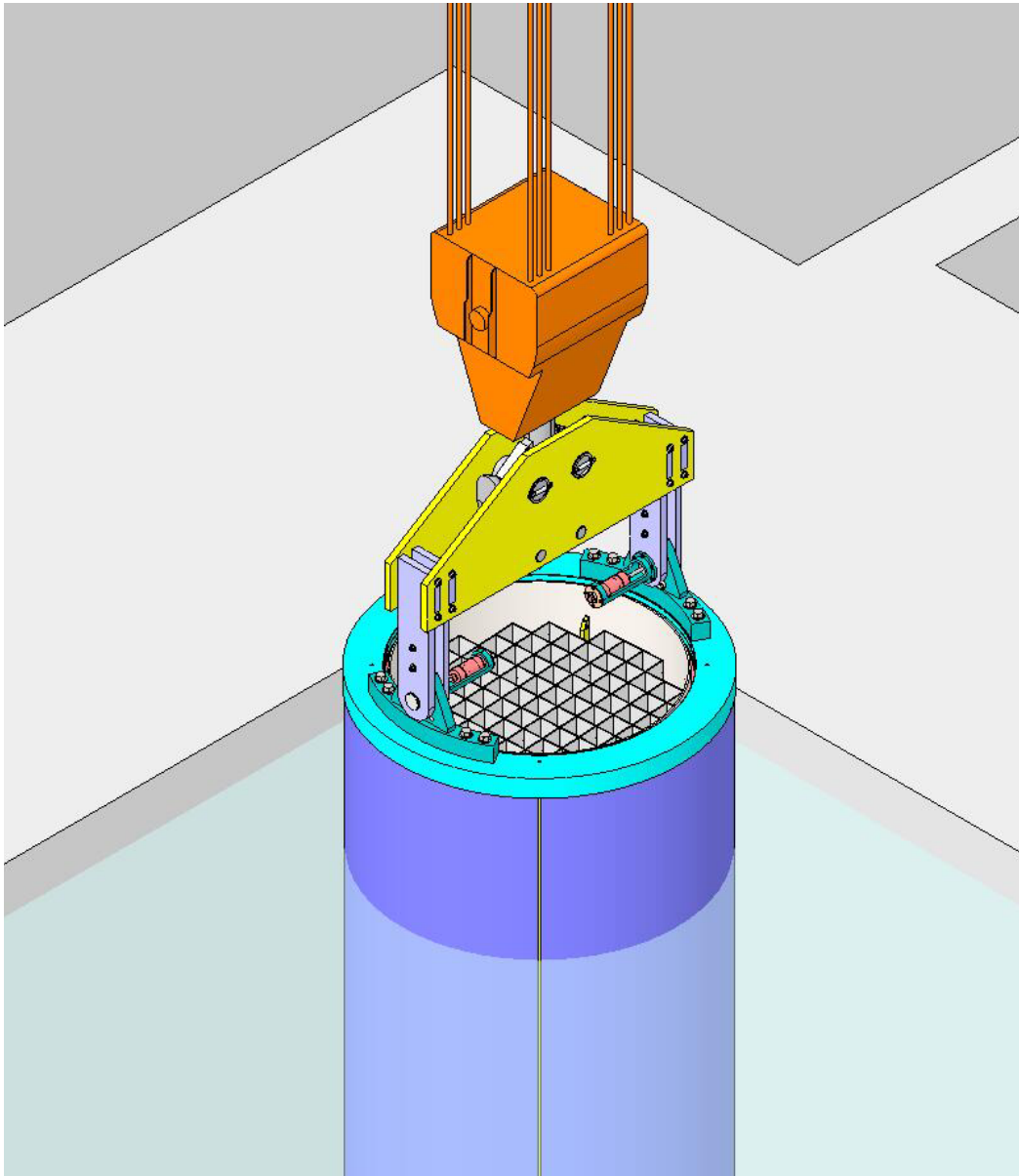
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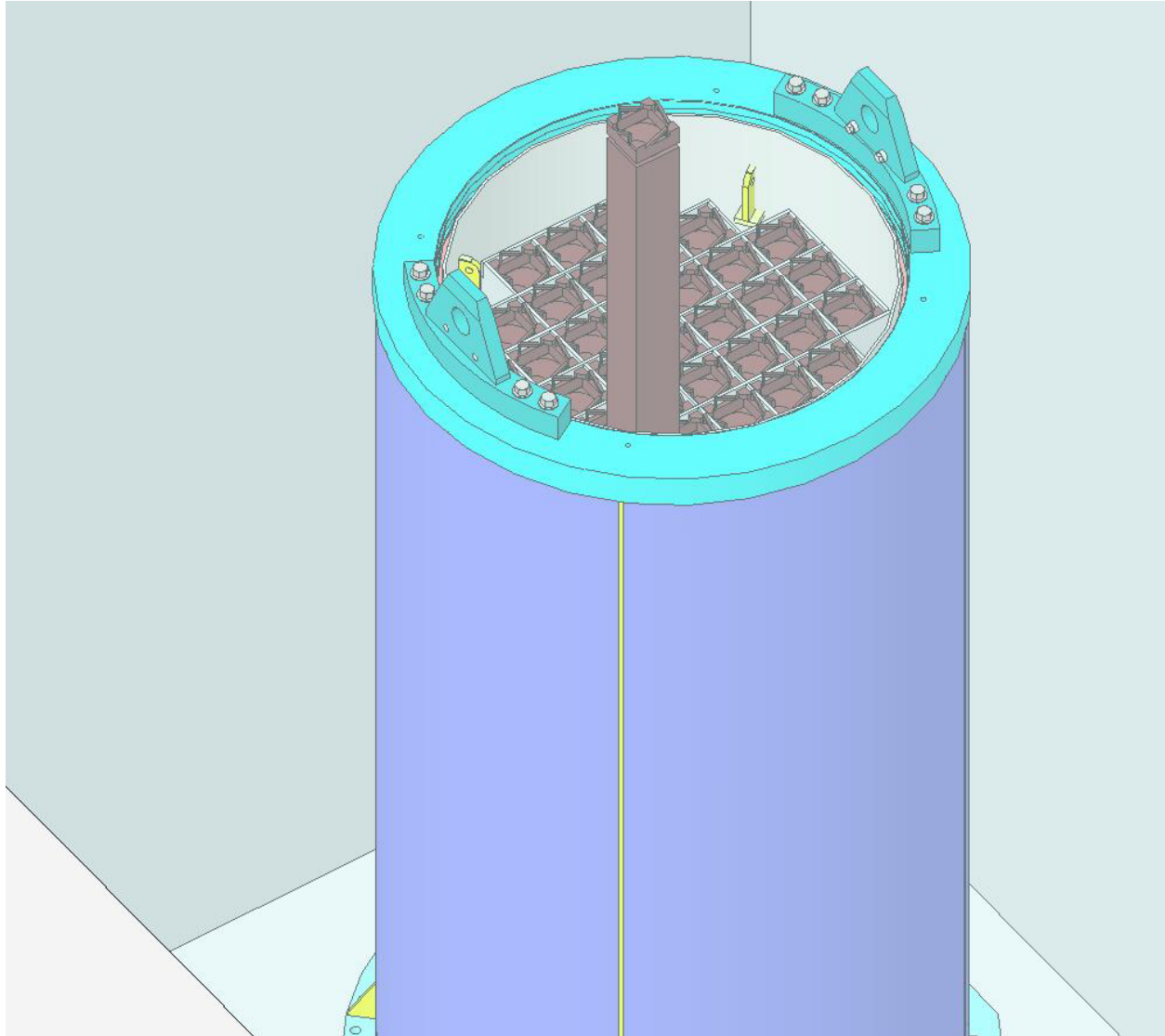
**FIGURE 9.2.1: MPC INSTALLATION IN HI-TRAC**



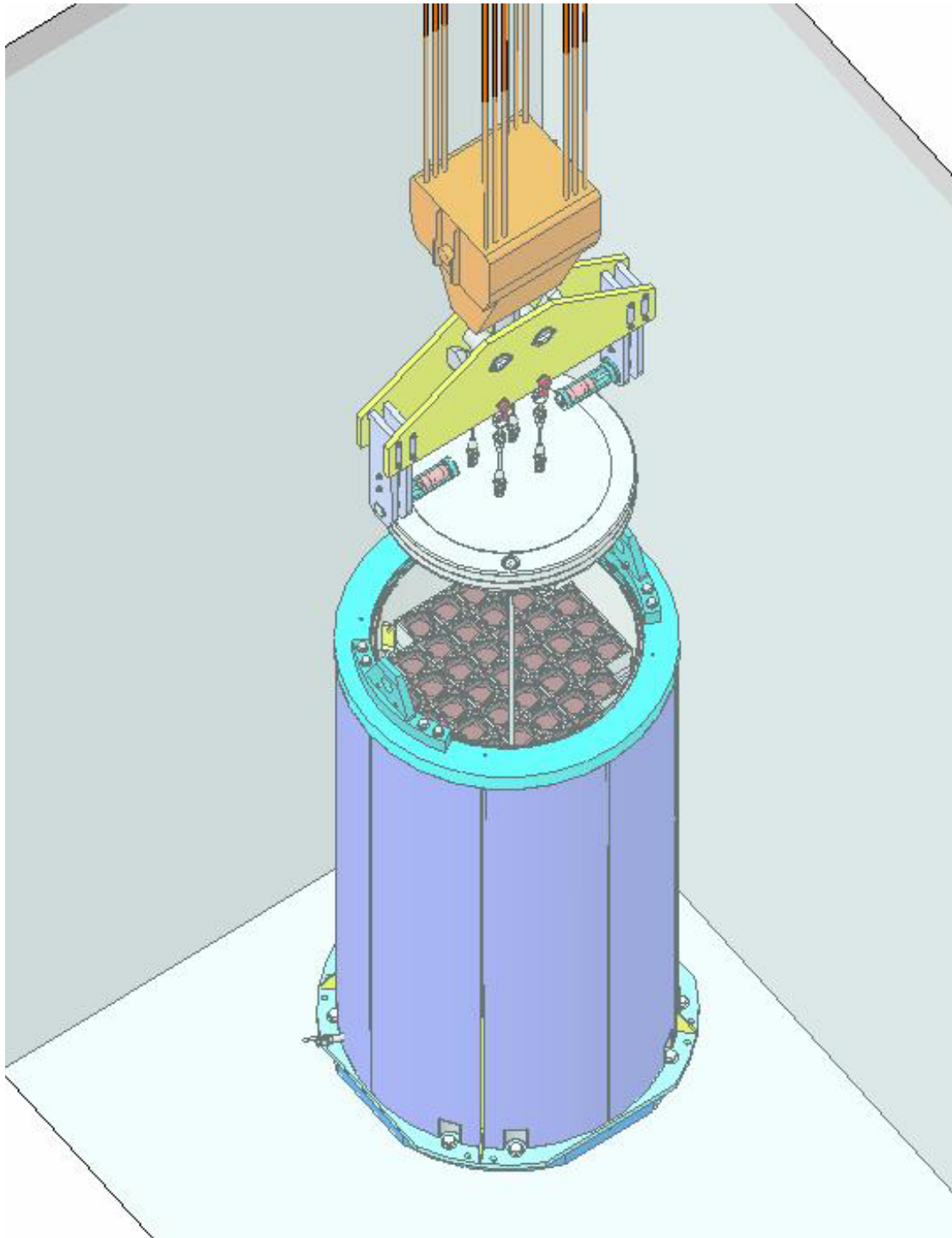
**FIGURE 9.2.2: HI-TRAC LIFTING SHOWN USING A REPRESENTATIVE LIFT YOKE**



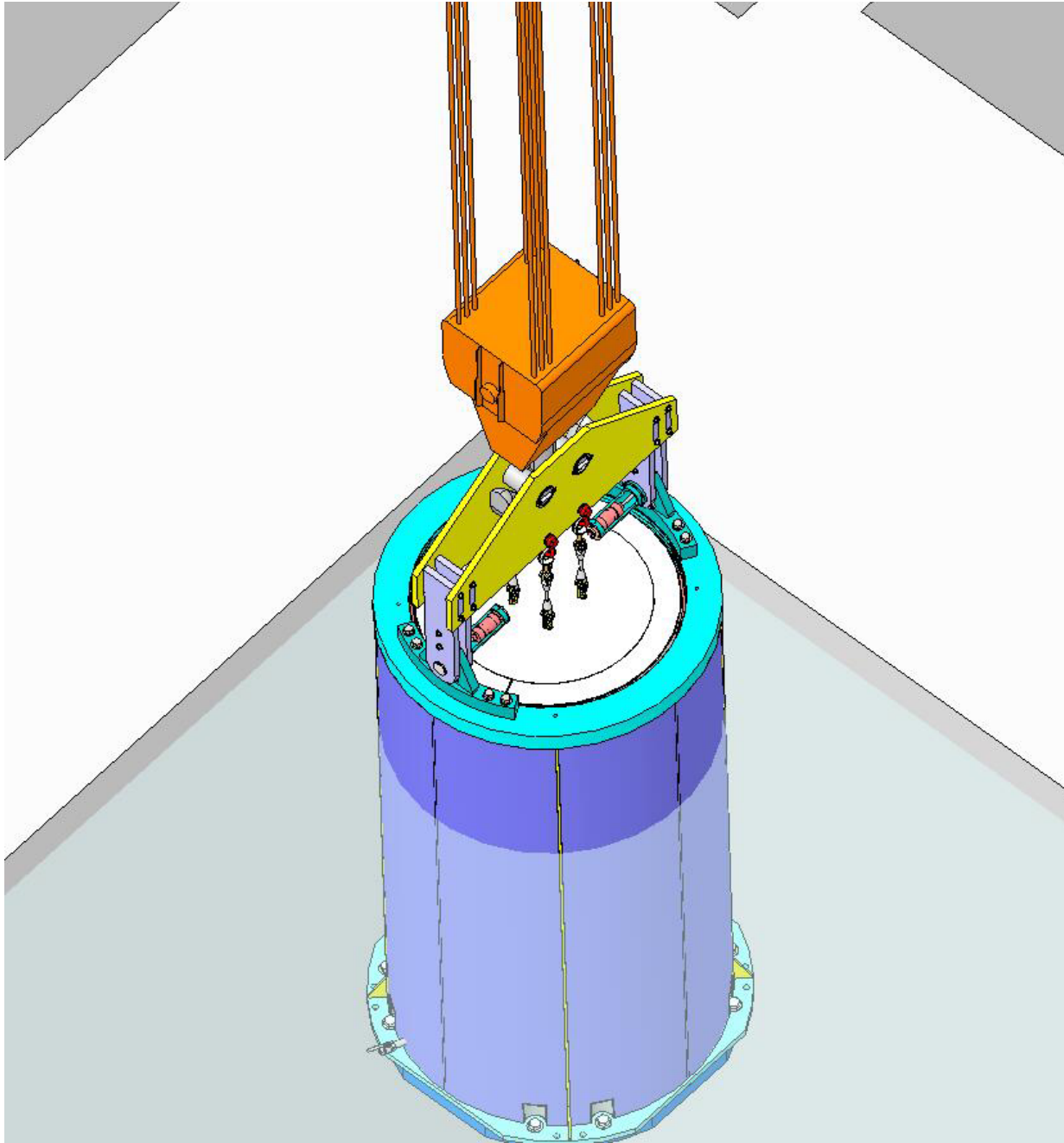
**FIGURE 9.2.3: HI-TRAC PLACEMENT IN THE SPENT FUEL POOL**



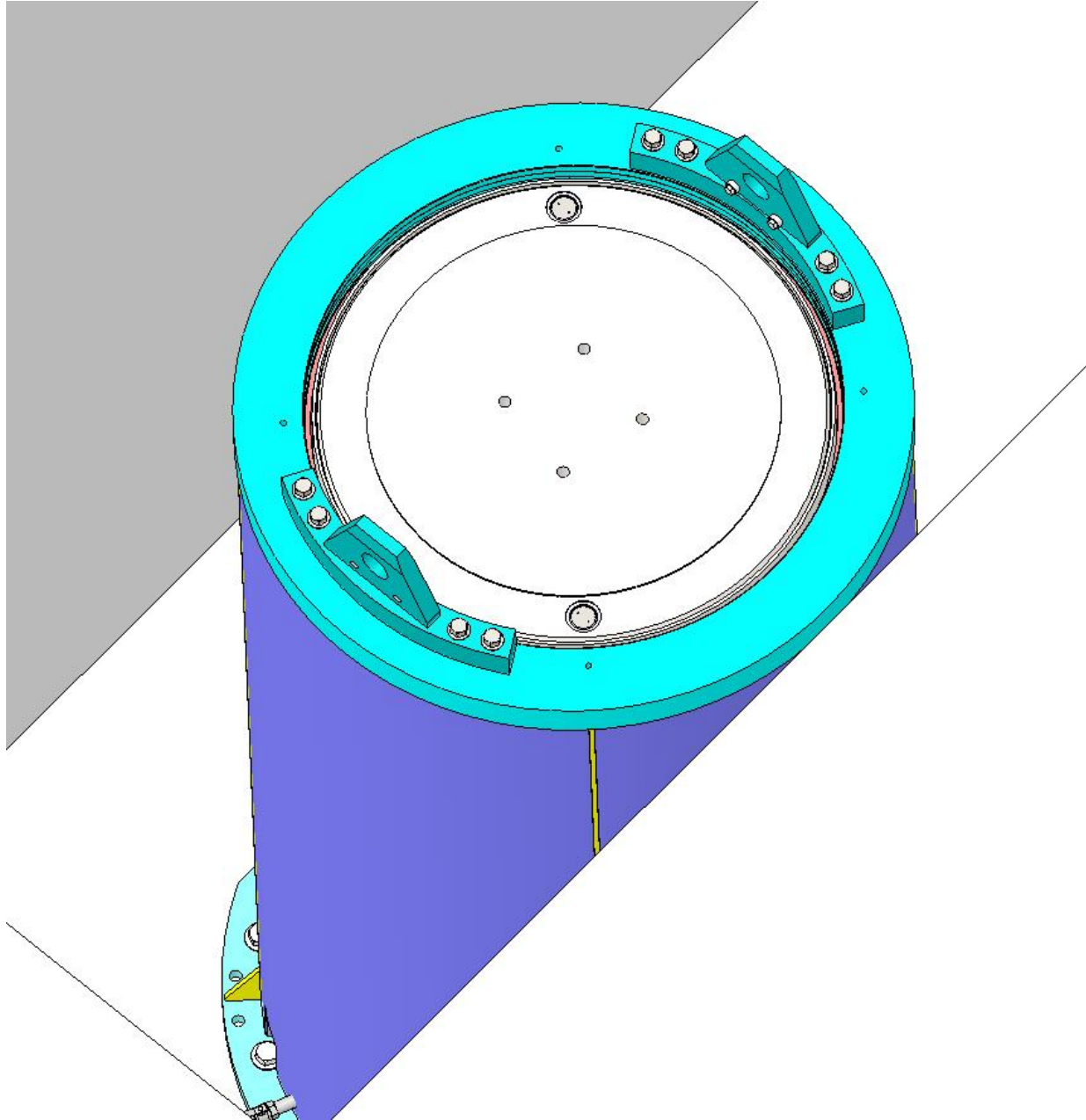
**FIGURE 9.2.4: FUEL ASSEMBLY PLACEMENT IN THE MPC  
(CRANE NOT SHOWN)**



**FIGURE 9.2.5: MPC LID INSTALLATION USING THE LIFT YOKE**

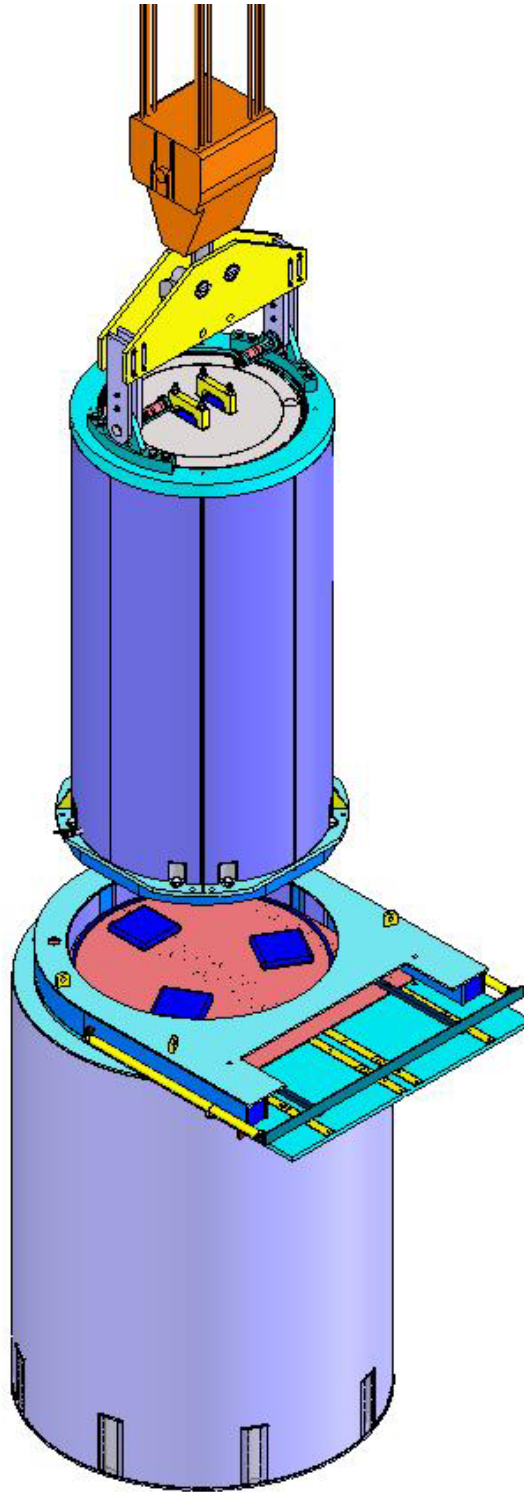


**FIGURE 9.2.6: HI-TRAC REMOVAL FROM THE SPENT FUEL POOL**

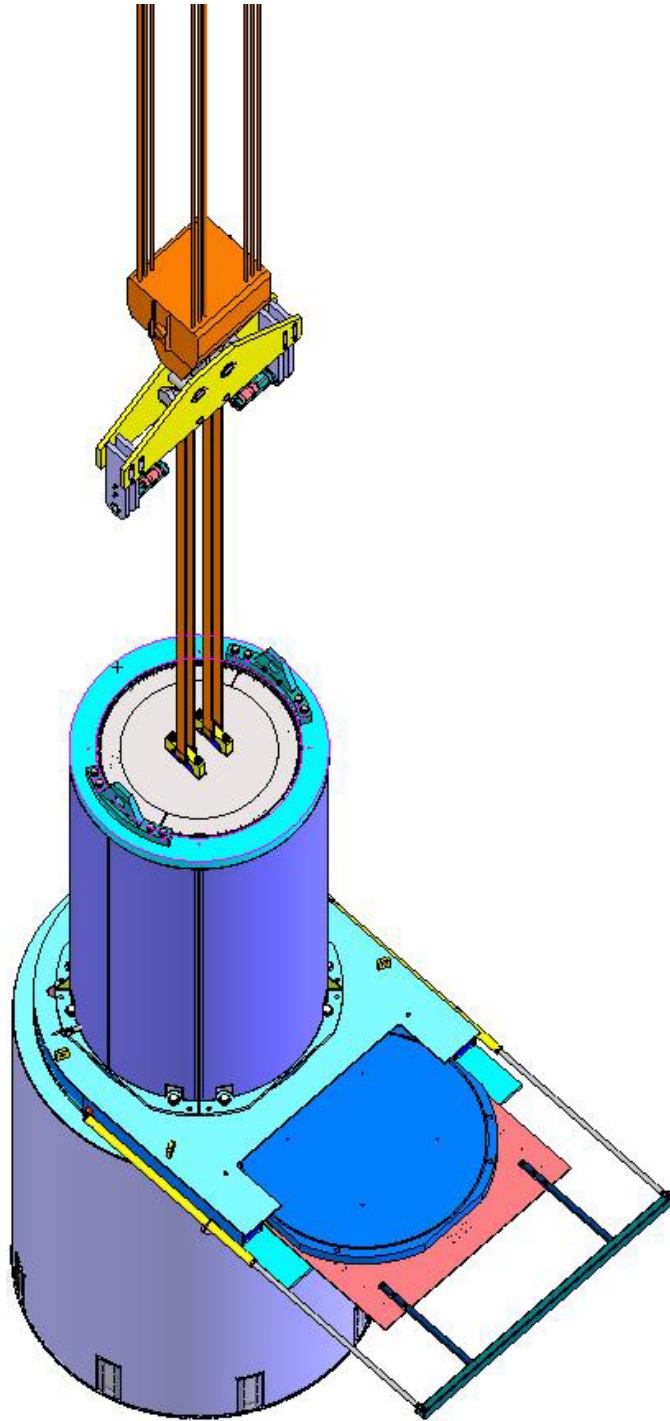


**FIGURE 9.2.7: HI-TRAC PLACEMENT IN THE CASK PREPARATION AREA**

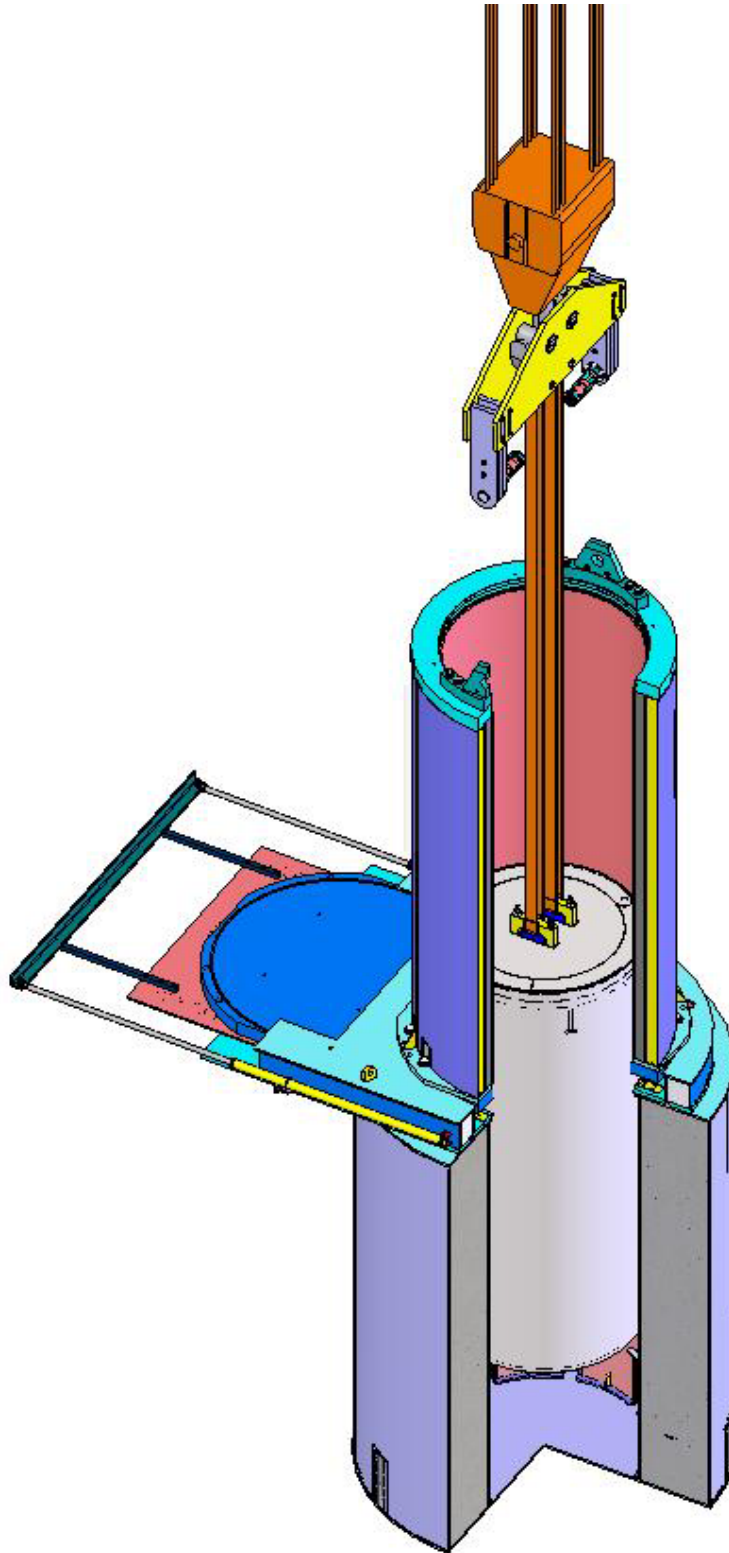




**FIGURE 9.2.8: HI-TRAC PLACEMENT ON THE HI-STORM 100 OVERPACK USING THE MATING DEVICE**



**FIGURE 9.2.9: HI-TRAC READY FOR MPC TRANSFER INTO  
HI-STORM FW OVERPACK**



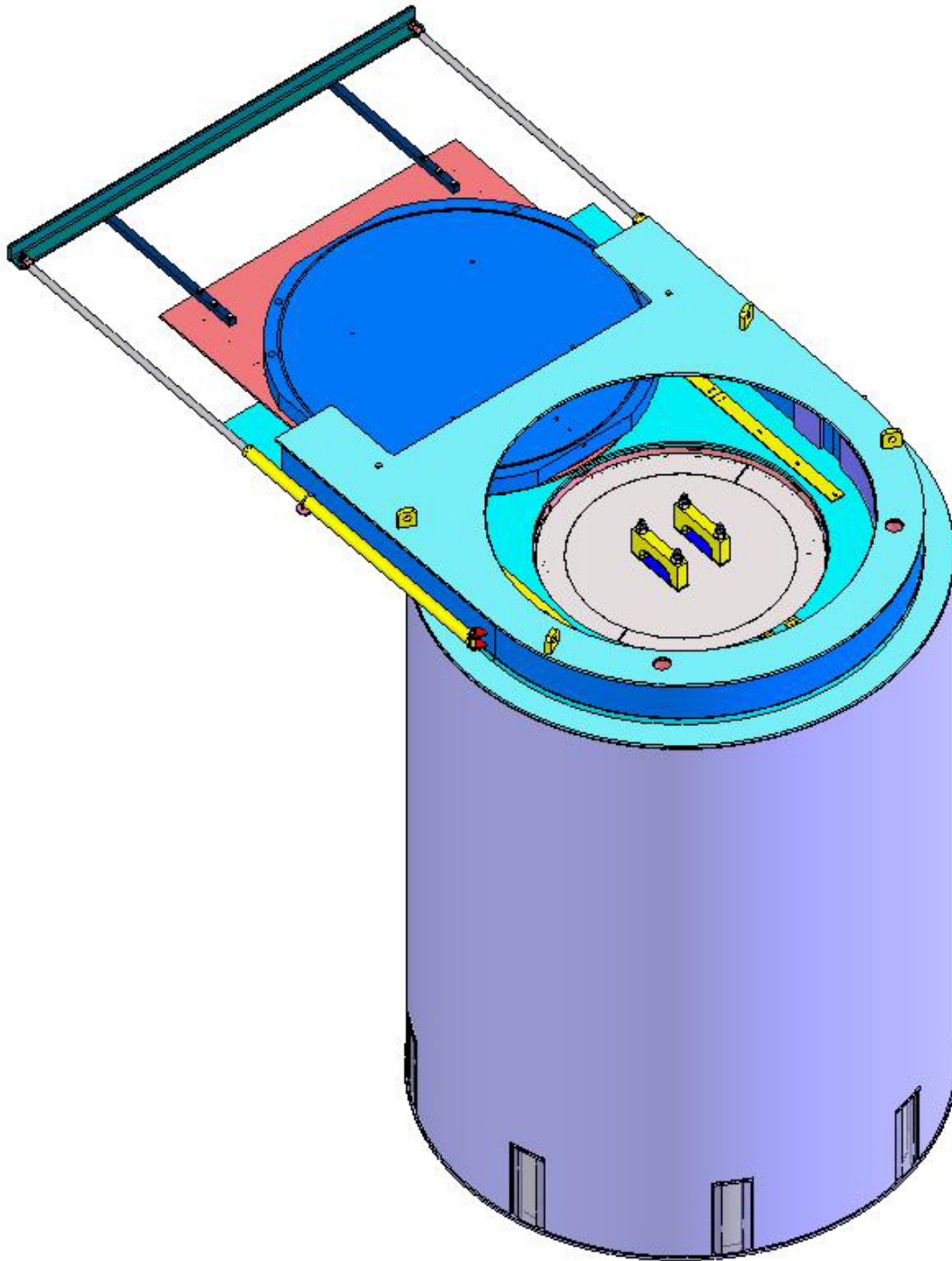
**FIGURE 9.2.10: MPC TRANSFER INTO HI-STORM FW OVERPACK  
(CUT-AWAY VIEW)**

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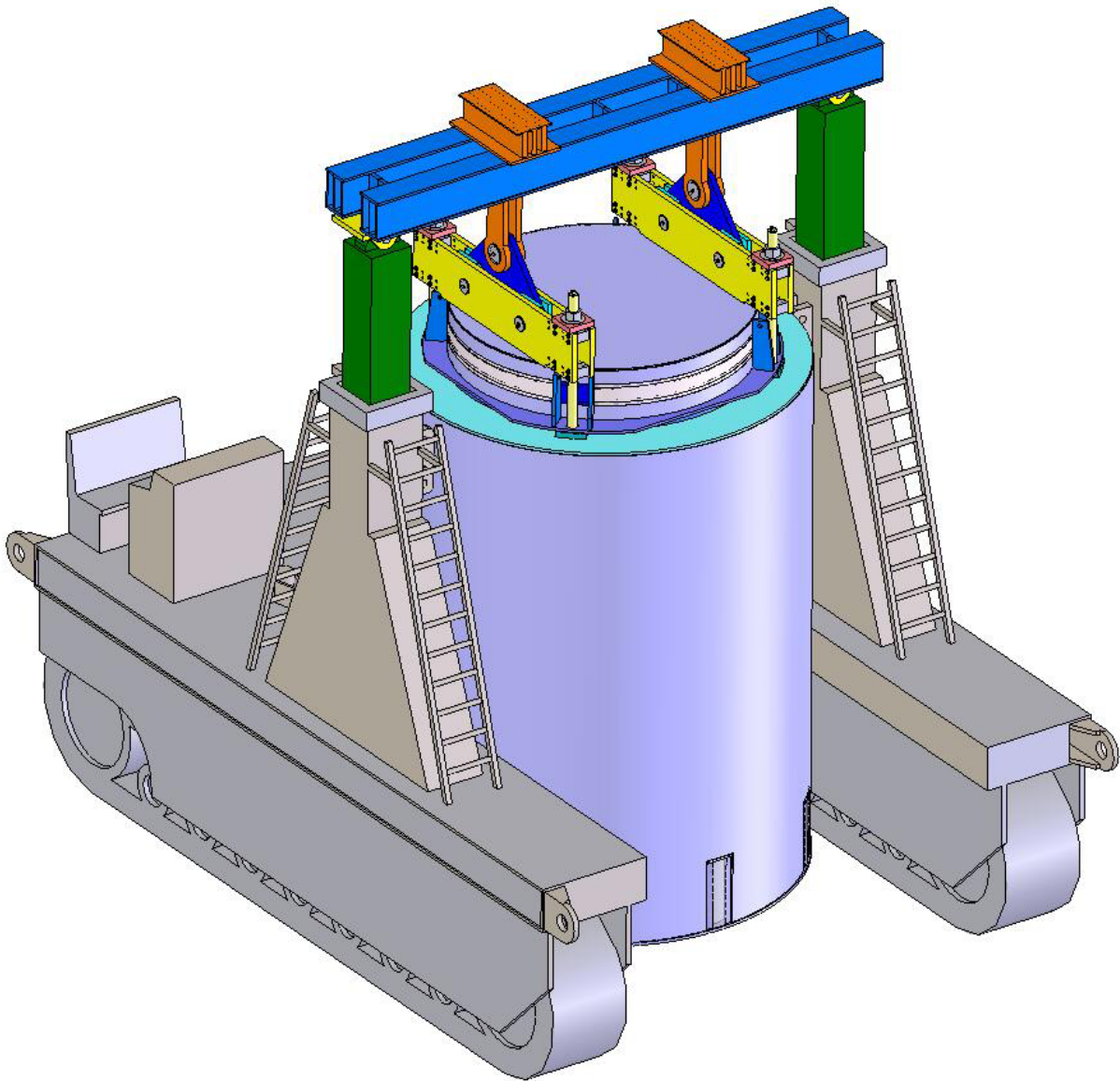
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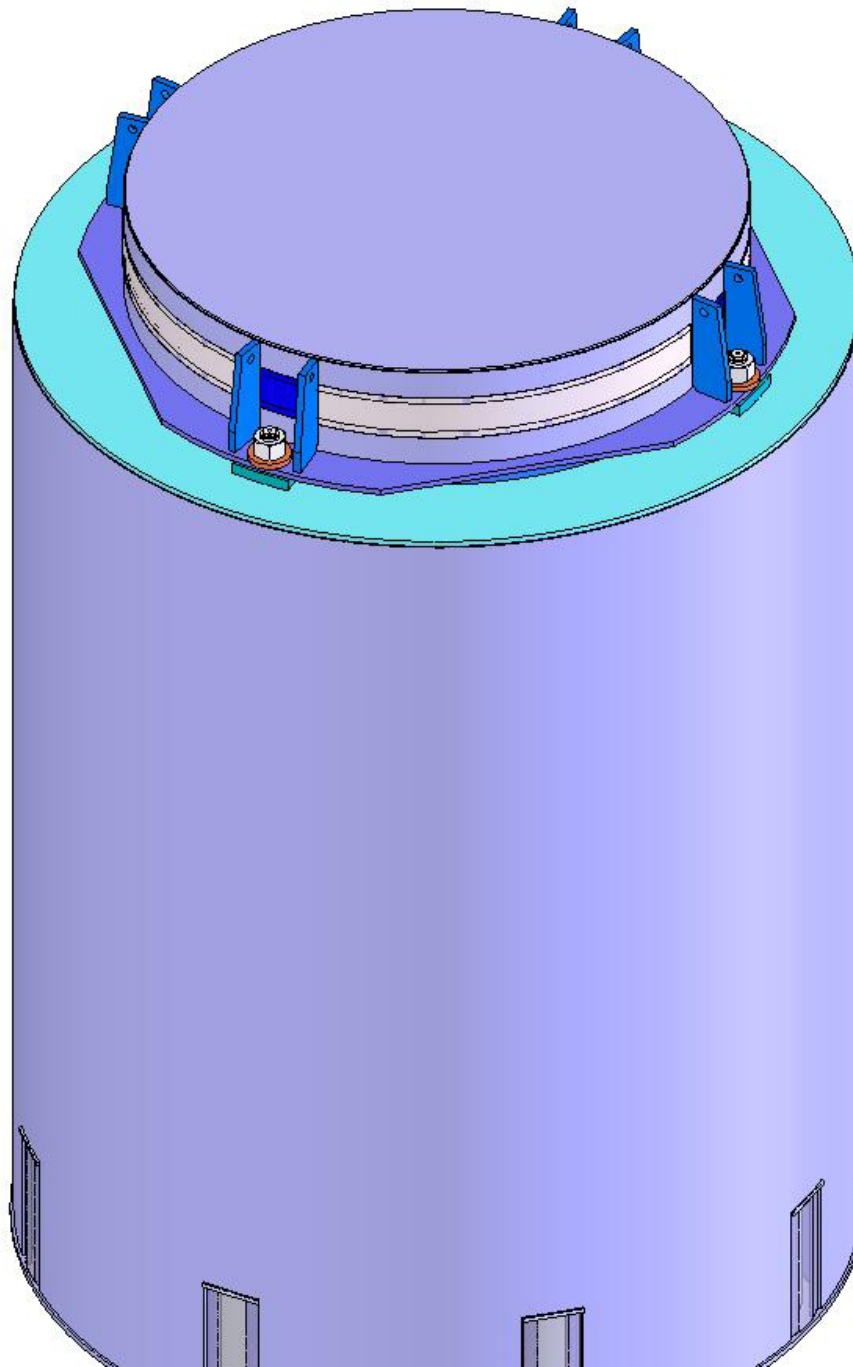
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**FIGURE 9.2.11: MPC SHOWN FULLY LOWERED INTO HI-STORM  
(HI-TRAC NOT SHOWN)**



**FIGURE 9.2.12: HI-STORM FW OVERPACK MOVEMENT SHOWN  
WITH A REPRESENTATIVE CASK TRANSPORTER**



**FIGURE 9.2.13: HI-STORM SHOWN IN STORAGE WITH THE LID  
INSTALLED**

### 9.3 ISFSI OPERATIONS

The HI-STORM FW system heat removal system is a totally passive system. Maintenance on the HI-STORM FW system is typically limited to cleaning and touch-up painting of the overpacks, repair and replacement of damaged vent screens, and removal of vent blockages (e.g., leaves, debris). The heat removal system operability surveillance should be performed after any event that may have an impact on the safe functioning of the HI-STORM FW system. These include, but are not limited to, wind storms, heavy snow storms, fires inside the ISFSI, seismic activity, flooding of the ISFSI, and/or observed animal or insect infestations. The responses to these conditions involve first assessing the dose impact to perform the corrective action (inspect the HI-STORM FW overpack, clear the debris, check the cask pitch, and/or replace damaged vent screens), perform the corrective action, verify that the system is operable (check ventilation flow paths and radiation). In the unlikely event of significant damage to the HI-STORM FW, the situation may warrant removal of the MPC, and repair or replacement of the damaged HI-STORM FW overpack. If necessary, the procedures in Section 9.2 may be used to reposition a HI-STORM FW overpack for minor repairs and maintenance. In extreme cases, Section 9.4 may be used as guidance for unloading the MPC from the HI-STORM FW.

**Note:**

The heat removal system operability surveillance involves performing a visual examination on the HI-STORM FW exit and inlet vent screens to ensure that the vents remain clear or verifying the temperature rise from ambient to outlet is within prescribed limits if using a temperature monitoring system. The metallic vent screens if damaged may allow leaves, debris, or animals to enter the duct and block the flow of air to the MPC.

**ALARA Warning:**

Operators should practice ALARA principles when inspecting the vent screens. Binoculars or boroscopes may be used to allow the operator to perform the surveillance from a low dose area.

1. Perform the heat removal operability surveillance in accordance with the CoC.
2. ISFSI Security Operations shall be performed in accordance with the approved site security program plan.



## 9.4 PROCEDURE FOR UNLOADING THE HI-STORM FW FUEL IN THE SPENT FUEL POOL

### 9.4.1 Overview of HI-STORM FW System Unloading Operations

**ALARA Note:**

The procedure described below uses the weld removal system to remove the welds necessary to enable the MPC lid to be removed. 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.

The HI-STORM FW system unloading procedures describe the general actions necessary to prepare the MPC for unloading, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover the HI-TRAC VW 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. The principal operational steps are summarized below.

The MPC is recovered from HI-STORM FW either at the ISFSI or the fuel building using the same methods as described in Section 9.2 (in reverse order). The HI-STORM FW lid is removed and the mating device is positioned on the HI-STORM FW. MPC slings are attached to the MPC lift attachment and positioned on the MPC lid. HI-TRAC VW is positioned on top of HI-STORM FW and the slings are brought through the top of the HI-TRAC VW. The MPC is raised into HI-TRAC VW, the mating device drawer is closed, and the bottom lid is bolted to the HI-TRAC VW. The HI-TRAC VW is removed from on top of HI-STORM FW.

HI-TRAC VW and its enclosed MPC are returned to the designated preparation area and the MPC lift rigging is removed. Water is added into the annulus space between the MPC and HI-TRAC VW, if required. The annulus and HI-TRAC VW top surfaces are covered to protect them from debris produced when removing the MPC lid weld. The weld removal system is installed and the MPC vent and drain ports are accessed. The vent RVOA is attached to the vent port and an evacuated sample bottle is connected. The vent port is slightly opened 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. A vent line is attached to the vent port and the MPC is vented to the fuel building ventilation system or spent fuel pool as determined by the site's radiation protection personnel. The MPC is filled with water (borated as required) at a controlled rate to avoid over-pressuring the MPC. The weld removal system then removes the MPC lid-to-shell weld. The weld removal system is removed with the MPC lid left in place.

The top surfaces of the HI-TRAC VW and MPC are cleared of metal shavings. The inflatable annulus seal is installed and pressurized. The MPC lid is rigged to the lift yoke and the lift yoke is engaged to HI-TRAC VW lift blocks. If weight limitations require, the neutron shield jacket is drained of water. HI-TRAC VW 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 cleared of

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any assembly debris and crud. HI-TRAC VW and MPC are returned to the designated preparation area where the MPC water is removed. The annulus water is drained and the MPC and overpack are decontaminated.

#### 9.4.2 HI-STORM FW Recovery from Storage

1. Recover the MPC from HI-STORM FW as follows:
  - a. Perform a transport route walkdown to ensure that the cask transport conditions are met.
  - b. Transfer HI-STORM FW to the fuel building or site designated location for the MPC transfer.
  - c. Position HI-STORM FW under the lifting device.
  - d. Remove the HI-STORM FW lid.
  - e. Install the mating device with bottom lid on top of the HI-STORM FW.
  - f. Remove the MPC lift attachment plugs and install the MPC lift rigging to the MPC lid.
2. At the site's discretion, perform a HI-TRAC VW receipt inspection and cleanliness inspection in accordance with a site-specific inspection checklist.

**Note:**

If the HI-TRAC VW is expected to be operated in an environment below 32 °F, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with demineralized water.

3. If previously drained, fill the neutron shield jacket with plant demineralized water or an ethylene glycol solution (25% ethylene glycol) as necessary. Ensure that the fill and drain plugs are installed.
4. Engage the lift yoke to HI-TRAC VW.
5. Align HI-TRAC VW over HI-STORM FW and mate the overpacks.
6. Unbolt the bottom lid and open the mating device drawer.
7. Attach the ends of the MPC sling to the lifting device.
8. Raise the MPC into HI-TRAC VW.
9. Verify the MPC is in the full-up position.

10. Close the mating device.
11. Bolt the bottom lid to the HI-TRAC VW.
12. Lower the MPC onto the bottom lid.
13. Disconnect the MPC lift rigging from the MPC lid.
14. Remove HI-TRAC VW from the top of the HI-STORM FW.

#### 9.4.3 Preparation for Unloading

1. Prepare for MPC cool-down as follows:

**Warning:**

At the start of annulus filling, the annulus fill water may flash to steam due to high MPC shell temperatures. Users may select the location and means of performing the annulus fill. Users may also elect the source of water for the annulus. Water addition should be performed in a slow and controlled manner until water steam generation has ceased.

2. If necessary, set the annulus water level to approximately 4 inches below the top of the MPC shell and install the annulus shield. Cover the annulus and HI-TRAC VW top surfaces to protect them from debris produced when removing the MPC lid weld.
3. Access the MPC as follows:

**ALARA Note:**

The following procedures describe weld removal using a machine tool head. Other methods may also be used. The metal shavings may need to be periodically removed.

**ALARA Warning:**

Weld removal may create an airborne radiation condition. Weld removal must be performed under the direction of the user's Radiation Protection organization.

- a. Using the marked locations of the vent and drain ports, core drill the closure ring and port cover plates.
- b. Remove the closure ring sections and the vent and drain port cover plates.

**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.

4. Take an MPC gas sample as follows:

**Note:**

Users may select alternate methods of obtaining a gas sample.

- a. Attach the RVOAs.
- b. Attach a sample bottle to the vent port RVOA.
- c. 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.

5. Fill the MPC cavity with water as follows:

**Caution:**

The MPC interior shall be filled with helium or another suitable inert gas to avoid exposing the fuel to oxidizing agents while at elevated temperatures. Exposing fuel at elevated temperatures to oxidizing agents can lead to deleterious oxidation of the fuel.

- a. Open the vent and drain port caps using the RVOAs.

**Caution:**

The introduction of water into the MPC may create water vapor. Re-flooding operations shall be closely controlled to ensure that the internal pressure in the MPC does not exceed design limits. The water flow rate shall be adjusted to maintain the internal pressure below design limits. See LCO 3.1.3 and SAR section 4.5.5.

**Caution:**

To mitigate unfavorable thermal shocking of the fuel cladding during re-flooding operations the re-flood water shall be at a temperature  $\geq 80^{\circ}\text{F}$ . See Section 3.4.4 for related fuel cladding evaluations.

- b. Route the vent port line several feet below the spent fuel pool surface or to the radwaste gas facility. Attach the vent line to the MPC vent port and slowly open the vent line valve to depressurize the MPC.

**Note:**

When unloading MPCs requiring soluble boron, the boron concentration of the water shall be checked in accordance with LCO 3.3.1 before and during operations with fuel and water in the MPC. Testing must be completed within four hours prior to unloading and every 48 hours after in accordance with the LCO until all the fuel is removed from the MPC. Two independent measurements shall be taken to ensure that the requirement of 10 CFR 72.124(a) is met.

- c. Attach the water fill line from a water source with water temperature  $\geq 80^{\circ}\text{F}$  to the MPC drain port and slowly open the water supply valve and establish a pressure less than 90 psi. (Refer to LCO 3.3.1 for boron concentration requirements). Fill the MPC until bubbling from the vent line has terminated. Close the water supply valve on completion.
- d. Disconnect both lines from the drain and vent ports leaving the drain port cap open to allow for thermal expansion of the water during MPC lid weld removal.

**Caution:**

A radiolysis of water may occur in high flux conditions inside the MPC creating combustible gases. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during MPC lid removal operations. The space below the MPC lid shall be purged with inert gas prior to, and during MPC lid removal operations, including grinding, and other hot work, to provide additional assurance that flammable gas concentrations will not develop in this space.

- e. 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 the gas space under the lid as necessary.
  - f. Remove the MPC lid-to-shell weld using the weld removal system.
  - g. Remove any metal shavings from the top surfaces of the MPC and HI-TRAC VW.
- 6. Install the inflatable annulus seal.
  - 7. Place HI-TRAC VW in the spent fuel pool as follows:
    - a. If necessary for plant weight limitations, drain the water from the neutron shield jacket.
    - b. Engage the lift yoke to HI-TRAC VW lifting blocks, remove the MPC lid lifting plugs and attach the MPC lid slings.

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- c. Position HI-TRAC VW into the spent fuel pool in accordance with site-approved rigging procedures.
- d. Disengage the lift yoke. Visually verify that the lift yoke is fully disengaged.
- e. Remove the lift yoke, MPC lid and drain line from the pool in accordance with directions from the site's Radiation Protection personnel.
- f. Disconnect the drain line from the MPC lid.
- g. Store the MPC lid components in an approved location. Disengage the lift yoke from MPC lid.

#### 9.4.4 MPC Unloading

- 1. Remove the spent fuel assemblies from the MPC using applicable site procedures.
- 2. Remove any debris or corrosion products from the MPC cells.

#### 9.4.5 Post-Unloading Operations

- 1. Remove HI-TRAC VW and the unloaded MPC from the spent fuel pool as follows:
  - a. Engage the lift yoke to the HI-TRAC VW lift blocks.
  - b. Apply slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the lift blocks.
  - c. Raise HI-TRAC VW until HI-TRAC VW flange is at the surface of the spent fuel pool.

**ALARA Warning:**

Activated debris may have settled on the top face of HI-TRAC VW during fuel unloading.

- d. Measure the dose rates at the top of HI-TRAC VW in accordance with plant radiological procedures and flush or wash the top surfaces to remove any highly-radioactive particles.
- e. Raise the top of HI-TRAC VW and MPC to the level of the spent fuel pool deck.
- f. Close the annulus overpressure system reservoir valve, if used.
- g. Lower the water level in the MPC approximately 12 inches to prevent splashing during cask movement.

**ALARA Note:**

To reduce contamination of HI-TRAC VW, the surfaces of HI-TRAC VW and lift yoke should be kept wet until decontamination can begin.

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- h. Remove HI-TRAC VW from the spent fuel pool under the direction of radiation protection personnel.
- i. Disconnect the annulus overpressure system from the HI-TRAC VW.
- j. Place HI-TRAC VW in the designated preparation area.
- k. Disengage the lift yoke.
- l. Perform decontamination on HI-TRAC VW and the lift yoke.
- m. Carefully decontaminate the area above the inflatable seal. Deflate, remove, and store the seal in an approved plant storage location.
- n. Using a water pump, pump the remaining water in the MPC to the spent fuel pool or liquid radwaste system.
- o. Drain the water in the annulus area by connecting the drain line to the HI-TRAC VW drain connector.
- p. Remove the MPC from HI-TRAC VW and decontaminate the MPC as necessary.
- q. Decontaminate HI-TRAC VW.
- r. Return any HI-STORM FW equipment to storage as necessary.

## 9.5 REFERENCES

- [9.0.1] U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems", NUREG-1536, Final Report, January 1997.
- [9.1.1] 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,"
- [9.1.2] 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.
- [9.1.3] American Society of Mechanical Engineers "Boiler and Pressure Vessel Code".
- [9.5.1] U.S. Code of Federal Regulations, Title 10 "Energy", Part 20, "Standards for Protection Against Radiation,"

# CHAPTER 10<sup>†</sup>: ACCEPTANCE CRITERIA AND MAINTENANCE PROGRAM

## 10.0 INTRODUCTION

This chapter identifies the fabrication, inspection, test, and maintenance programs to be conducted on the HI-STORM FW system (overpack, MPC and transfer cask) 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 (CoC). The acceptance criteria and maintenance program requirements specified in this chapter apply to each HI-STORM FW system fabricated, assembled, inspected, tested, and accepted for use under the purview of the HI-STORM FW 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-STORM FW system will maintain confinement of radioactive material under normal, off-normal, and hypothetical 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-STORM FW system operations to assure that the HI-STORM FW system is functioning within its design parameters. These include receipt inspections, nondestructive weld examinations, pressure tests, radiation shielding tests, thermal performance tests, dryness tests, and others. Chapter 9 identifies the tests and inspections. "Pre-operation" as referred to in this chapter defines that period of time from receipt inspection of a HI-STORM FW system until the empty MPC is loaded into a HI-TRAC transfer cask for fuel assembly loading.

The HI-STORM FW system is classified as important-to-safety. Therefore, the individual structures, systems, and components (SSCs) that make up the HI-STORM FW 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-STORM FW system.

The acceptance criteria and maintenance program described in this chapter comply with the requirements of 10CFR72 [10.0.1] and NUREG-1536 [10.0.2] to the maximum extent possible, as described in Chapter 1.

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<sup>†</sup> 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. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the Glossary.

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The acceptance test requirements on the manufactured welds in the HI-STORM FW system are contained in the component licensing drawings in Section 1.5. Additional details on the requirements in the drawings 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.

## 10.1 ACCEPTANCE CRITERIA

This section provides the workmanship inspections and acceptance tests to be performed on the HI-STORM FW system prior to and during loading of the system. These inspections and tests provide the assurance that the HI-STORM FW 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 provisions of 10CFR72 [10.0.1].

The testing and inspection acceptance criteria applicable to the MPCs, the HI-STORM FW overpack, and the HI-TRAC VW transfer casks are listed in Tables 10.1.1, 10.1.2, and 10.1.3, respectively, and discussed in more detail in the sections that follow. Chapters 9 and 13 provide operating guidance and the bases for the Technical Specifications, respectively. These inspections and tests are intended to demonstrate that the HI-STORM FW system has been fabricated, assembled, and examined in accordance with the design criteria contained in Chapter 2 of this FSAR. Identification and resolution of manufacturing noncompliances, if any, shall be performed in accordance with the Holtec International Quality Assurance Program approved by the USNRC.

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.

### 10.1.1 Fabrication and Nondestructive Examination (NDE)

This subsection summarizes the test program required for the HI-STORM FW system.

#### 10.1.1.1 Fabrication Requirements

The following fabrication controls and required inspections shall be performed on the HI-STORM FW system, including the MPCs, overpacks, and HI-TRAC transfer casks, in order to assure compliance with this FSAR and the Certificate of Compliance.

- i. Materials of construction specified for the HI-STORM FW system are identified in the drawings in Chapter 1 and shall be procured with certification and supporting documentation as required by the ASME Code [10.1.1] Section II (where applicable),

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the requirements of ASME Section III (where applicable), Holtec procurement specifications, and 10CFR72, Subpart G. Materials and components shall be receipt inspected for visual and dimensional acceptability, material conformance to specification requirements, and traceability markings, as applicable. Controls shall be in place to ensure that material traceability is maintained throughout fabrication. Materials for the Confinement Boundary (MPC baseplate, lid, closure ring, port cover plates and shell) shall also be procured in compliance with the requirements of ASME Section III, Article NB-2500.

- ii. The MPC Confinement Boundary shall be fabricated and inspected in accordance with ASME Code, Section III, Subsection NB to the extent practicable, as explained in this chapter.
- iii. ASME Code welding shall be performed using welders and weld procedures that have been qualified in accordance with ASME Code Section IX and the applicable code (such as ASME Section III Subsection NB for the Confinement Boundary).
- iv. Code welds shall be visually examined in accordance with ASME Code, Section V, Article 9. The acceptance criteria for the welds shall be based on the ASME Codes provided in Table 10.1.5. These additional NDE criteria are also specified on the licensing drawings in Section 1.5 for the specific welds. Weld inspections shall be detailed in a weld inspection plan which shall identify the weld and the examination requirements, the sequence of examination, and the acceptance criteria. The inspection plan shall be subject to review and approval by Holtec in accordance with the Company's QA program prior to use. NDE inspections of code welds shall be performed in accordance with written and approved procedures by personnel qualified in accordance with SNT-TC-1A [10.1.2] or other site-specific, NRC-approved program for personnel qualification.
- v. The MPC confinement 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.
- vi. Repair of confinement boundary welds shall conform to the requirements of the ASME Code, Section III, Article NB-4450.
- vii. Base metal repairs shall be performed and examined in accordance with the applicable reference code set down in Table 10.1.5.
- viii. Grinding and machining operations on the MPC Confinement Boundary shall be controlled through written and approved procedures to ensure grinding and machining operations do not reduce local base metal wall thicknesses of the Confinement Boundary below allowable limits. The thicknesses of base metals shall be ultrasonically tested, as necessary, in accordance with written and approved

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procedures to verify base metal thickness meets the applicable requirements.

- ix. 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 an integral part of a permanent weld shall be examined by an approved visual examination procedure.
- x. The HI-STORM FW system shall be inspected for cleanliness and proper packaging for shipping in accordance with written and approved procedures.
- xi. Each cask shall 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.
- xii. A documentation package shall be prepared and maintained during fabrication of each HI-STORM FW system to include detailed records and evidence that the required inspections and tests have been performed. The completed documentation package shall be reviewed to verify that the HI-STORM FW system or component has been properly fabricated and inspected in accordance with the design and Code construction requirements. The documentation package shall include, as applicable, but not be limited to:
  - Completed Shop Weld Records
  - Inspection Records
  - Nonconformance Reports
  - Material Test Reports
  - NDE Reports
  - Dimensional Inspection Report

#### 10.1.1.2 MPC Lid-to-Shell Weld Inspection

- i. The MPC lid-to-shell (LTS) weld shall be examined using a progressive multi-layer liquid penetrant (PT) examination during welding.
- ii. The multi-layer PT must, at a minimum, include the root and final weld layers and one intermediate PT after each approximately 3/8 inch weld depth has been completed as specified in the drawing package in Section 1.5.

The inspection results, including relevant 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.

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The multi-layer PT examination of the LTS weld, in conjunction with other examinations and tests performed on this weld, shall use ASME Section III acceptance criteria (see Table 10.1.4) which provide reasonable assurance that the LTS weld is sound and will perform its design function under all loading conditions. The multi-layer PT examination and evaluation of indications provides reasonable assurance that leakage of the weld or structural failure under the design basis normal, off-normal, and accident loading conditions will not occur.

#### 10.1.1.3 Visual Inspections and Measurements

The HI-STORM FW 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 approved in NRC Docket No. 71-0784.

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-STORM FW system shall become part of the final quality documentation package.

#### 10.1.1.4 Weld Examination

The examination of the HI-STORM FW 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, although they are conservatively not credited in structural analysis, shall be examined and repaired in accordance with NDE specified in the drawing package and with written and approved procedures developed specifically for welding Metamic-HT with acceptance criteria per ASME Section V, Article 1, Paragraph T-150 (2007 Edition). The basket welds, made by the Friction Stir Weld process, are classified as Category C per NG-3351.3 and belonging to Type III (by virtue of being corner joint with a thru-thickness "stir zone") in Table NG-3352-1. These weld requirements are not applicable to welds identified as NITS on the drawing package.
3. Non-code welds shall be examined and repaired in accordance with written and approved procedures as defined in the system Manufacturing Manual.

## 10.1.2 Structural and Pressure Tests

### 10.1.2.1 Lifting Locations

The HI-STORM FW system does not utilize any lifting trunnions. The lifting of all HI-STORM FW components is engineered to occur through threaded couplings integral to the strongest part in the component. Thus, as shown in the HI-TRAC VW drawings (Section 1.5) the threaded connection is located in the top forging. These lift locations are accordingly referred to as *tapped anchor locations* (TAL). The TALs to lift the MPCs (in all Holtec designs) is located in the top lid (thickest part) and those for the HI-STORM FW overpack are welded to the radial connector plates (in all HI-STORM models).

Because the TALs are integral to the component, they possess high ductility and shall meet the requirements as shown in Paragraph 3.4.3.1 and Table 2.2.6.

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 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 load drop accidents by implementing measures to eliminate shortcomings in all aspects of the operation including the four aforementioned areas.

Each TAL will be subjected to a dimensional test in the shop using go/no-go gauges to ensure that the threads meet the dimensional requirements. As an alternative to the thread gauge test, the threads may be proof-tested using a torque test to simulate a load equal to three times the design load. Furthermore, the thread in the TAL shall be visually inspected in accordance with a written procedure to ensure absence of burrs, undercuts, and other stress raisers.

The acceptance testing of the TALS in the manner described above will provide adequate assurance against handling accidents.

#### 10.1.2.2 Pressure Testing

##### 10.1.2.2.1 HI-TRAC Transfer Cask Water Jacket

All HI-TRAC transfer cask water jackets shall be hydrostatically tested in accordance with written and approved procedures. The water jacket fill port will be used for filling the cavity with water and the vent port for venting the cavity. The approved test procedure shall clearly define the test equipment arrangement.

The hydrostatic test shall be performed after the water jacket has been welded together. The test pressure gage installed on the water jacket shall have an upper limit of approximately twice that of the test pressure. The hydrostatic test pressure shall be maintained for ten minutes. During this time period, the pressure gage shall not fall below the applicable minimum test pressure. At the end of ten minutes, and while the pressure is being maintained at the minimum pressure, weld joints shall be visually examined for leakage. If a leak is discovered, the cavity shall be emptied and an examination to determine the cause of the leakage shall be made. Repairs and retest shall be performed until the hydrostatic test criteria are met.

After completion of the hydrostatic testing, the water jacket exterior surfaces shall be visually examined for cracking or deformation. Evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable. Unacceptable areas shall require repair and re-examination per the applicable ASME Code. The HI-TRAC water jacket hydrostatic test shall be repeated until all examinations are found to be acceptable.

Test results shall be documented. The documentation shall become part of the final quality documentation package.

##### 10.1.2.2.2 MPC Confinement Boundary

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

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welding of the MPC lid-to-shell weld is completed. If hydrostatic testing is used, the MPC shall be pressure tested to 125% of design pressure. If pneumatic testing is used, the MPC shall be pressure tested to 120% of design pressure. 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 written and approved procedures prepared in accordance with the ASME Code, Section III, Article NB-4450.

The MPC confinement boundary pressure test shall be repeated until all required examinations are found to be acceptable. Test results shall be documented and maintained as part of the loaded MPC quality documentation package.

### 10.1.3 Materials Testing

The majority of materials used in the HI-TRAC transfer cask and a portion of the material in the HI-STORM 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.

Materials of the HI-TRAC transfer cask and HI-STORM overpack, as required, shall be Charpy V-notch tested in accordance with ASME Section IIA and/or ASME Section III, Subsection NF, Articles NF-2300, and NF-2430. The materials to be tested are identified in Table 3.1.9 and applicable weld materials. Table 3.1.9 provides the test temperatures and test acceptance criteria to be used when performing the material testing specified above.

The concrete utilized in the construction of the HI-STORM overpack shall be mixed, poured, and tested as set down in Chapter 1.D of the HI-STORM 100 FSAR (Docket 72-1014) [10.1.6] in accordance with written and approved procedures. Testing shall verify the compressive strength and density meet design requirements. Tests required shall be performed at a frequency as defined in the applicable ACI code.

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Qualification tests on Metamic-HT coupons drawn from production runs shall be performed in compliance with Table 10.1.6 requirements to ensure that the manufactured panels shall render their intended function. Testing shall be performed using written and approved procedures consistent with the test methods documented in Holtec's test report [10.1.7]. To ensure the above test requirements are met a sampling plan based on the MIL Standard 105E [10.1.8] is defined and incorporated in the Metamic-HT Manufacturing Manual's Shop Operating Procedure HTSOP-108.

Test results on all materials shall be documented and become part of the final quality documentation package.

#### 10.1.4 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 [10.1.5], 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 "leaktight" as defined in ANSI N14.5. 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 not required. 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. The description and procedures for these field leakage tests are provided in Chapter 9 of this FSAR and the acceptance criteria are defined in the Technical Specifications for the HI-STORM FW system.

#### 10.1.5 Component Tests

##### 10.1.5.1 Valves, Pressure Relief Devices, and Fluid Transport Devices

There are no fluid transport devices associated with the HI-STORM FW system. The only valve-like components in the HI-STORM FW system are the specially 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.

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There are multiple pressure relief devices installed in the upper ledge surface of the HI-TRAC transfer cask water jacket. One is provided for venting air and water due to pressure build-up from thermal expansion of the water in the water jacket. The other relief devices are provided for venting of the neutron shield jacket fluid under hypothetical fire accident conditions in which the design pressure of the water jacket may be exceeded. The set pressures for the pressure relief devices are listed on the HI-TRAC VW drawings in Section 1.5.

#### 10.1.5.2 Seals and Gaskets

There are no confinement seals or gaskets included in the HI-STORM FW system.

### 10.1.6 Shielding Integrity

The HI-STORM FW overpack and MPC have two designed shields for neutron and gamma ray attenuation. The HI-STORM FW overpack concrete provides both neutron and gamma shielding. The overpack's inner and outer steel shells, and the steel shield shell, provide radial gamma shielding. Concrete and steel plates provide axial neutron and gamma shielding.

The HI-TRAC VW transfer cask uses three different materials for primary shielding. All HI-TRAC VW transfer cask designs include a radial steel-lead-steel shield and a removable steel bottom lid. Testing requirements on shielding materials are presented below.

#### Concrete:

The dimensions of the HI-STORM overpack steel shells and the density of the concrete shall be verified to be in accordance with FSAR drawings in Section 1.5 prior to concrete installation. The dimensional inspection and density measurements shall be documented. Also, see Subsection 10.1.3 for concrete material testing requirements.

#### Lead:

The installation of the lead in the HI-TRAC transfer cask shall be performed using written and qualified procedures in order to ensure that voids are minimized. The lead shall be examined to preclude macrovoids (through holes) in the material using written and qualified procedures.

The lead shall be installed in such a manner that there are no macro-voids (through holes) and that the cask is not subjected to a severe thermal cycle.

#### Steel:

Steel plates utilized in the construction of the HI-STORM FW system shall be dimensionally inspected to assure compliance with the requirements specified on the Design Drawings.

#### General Requirements for Shield Materials:

1. Test results for concrete density and lead examinations for macrovoids, as applicable, shall be documented and become part of the quality documentation package.
2. Dimensional inspections of the cavities containing the shielding materials shall assure that the design required amount of shielding material is being incorporated into the fabricated item.

Shielding effectiveness tests shall be performed after initial loading operations in accordance with description below and the operating procedures in Chapter 9.

#### 10.1.6.1 Shielding Effectiveness Tests

Operational neutron and gamma shielding effectiveness tests shall be performed after fuel loading using written and approved procedures at the host plant site. Calibrated neutron and gamma dose rate meters shall be used to measure the actual neutron and gamma dose rates at the surface of the HI-STORM FW overpack and HI-TRAC VW. 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 carried out. Dose rate measurements shall be documented and shall become part of the quality record of the loaded cask.

#### 10.1.6.2 Neutron Absorber Manufacturing Requirements

Essential characteristics of Metamic-HT are described in Chapter 1 of this FSAR. As described in Chapter 1, Metamic-HT is made from high purity aluminum using a powder metallurgy process that results in pinning of the materials grain boundaries by dispersoids of nanoparticles of aluminum oxide. The manufacturing of Metamic-HT is governed by a set of quality validated Holtec Standard procedures contained in the Metamic-HT Manufacturing Manual [1.2.7].

The key constituents of Metamic-HT, namely aluminum powder and Boron Carbide powder are procured under their respective purchasing specifications that define the required particle size distributions and set down the prohibited materials & impurities, as well as tolerable level of impurities. The supplier of raw materials must be qualified under Holtec's quality program for important to safety materials and components or the material shall be commercially dedicated by Holtec in accordance with the Holtec Quality Assurance program.

A description of the manufacturing processes for Metamic-HT is presented in the Metamic-HT Sourcebook [1.2.6] and implemented in the Metamic-HT Manufacturing Manual [1.2.7].

As required by the procedures set down in its manufacturing manual [1.2.6], each panel of Metamic-HT neutron absorber material shall be visually inspected for damage such as scratches, cracks, burrs, presence of imbedded foreign materials, voids and discontinuities that could significantly affect its functional effectiveness.

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Metamic-HT panels will be manufactured according to a Holtec purchase specification that incorporates all requirements set forth in this FSAR. The manufacturing of Metamic-HT is subject to all quality assurance requirements under Holtec International's NRC approved quality program.

The tests conducted on Metamic-HT to establish the compliance of the manufactured panels with Holtec's Purchasing Specification are intended to ensure that *critical characteristics* of the final product will meet the minimum guaranteed values (MGVs) set forth in this FSAR (Table 1.2.8a). The tests are performed at both the raw material and manufactured extrusion/panel stages of production with the former serving as the insurer of the properties in the final product and the latter serving the confirmatory function.

The testing is conducted for each lot of raw material and finished panels as prescribed in Table 10.1.6. A lot is defined as follows:

“Lot” means a population of an item that shares identical attributes that are central to defining a critical performance or operational characteristic required of it. Thus, a lot of boron carbide powder procured to a certified Purchasing Specification used in the manufacturing of Metamic-HT is the bulk quantity of the powder that has the same particle size distribution. A lot of finished panels drawn from a powder mix and manufactured in an extrusion run have identical aluminum and boron carbide characteristics and the same extrusion conditions.

The following tests are performed (see Table 10.1.6):

(i) Testing and certification of powder material

- All lots of aluminum and boron carbide powder shall be certified to meet particle size distribution and chemistry requirements in [1.2.8].
- All lots of B<sub>4</sub>C shall be certified as containing Boron with the minimum isotopic B-10 per the boron carbide purchase specifications incorporated in the Manufacturing manual [1.2.7].
- Homogenized mixtures of Al powder(s) and boron carbide powder(s) from traceable lots, prepared for sintering and billet forming operations, shall have the minimum boron carbide wt% verified by wet chemistry testing of one sample from each lot of blended powders. The mixing/blending of the batch shall be controlled via approved procedures.

(ii) Testing of finished panels

The number of panels subject to testing shall be governed by Table 10.1.7. The panels that need to be tested per the statistical protocol of Table 10.1.7, hereafter referred to as test panels, shall be subject to the following evaluations:

- The Metamic-HT panels shall be tested for all mechanical properties in Table 1.2.8a in accordance with Table 10.1.7 sampling plan.
- The thickness of each panel will be measured using the procedure set down in [1.2.7]. The average measured value must meet the minimum basket wall requirements specified in the Licensing drawings in Section 1.5.
- One coupon from the test panel shall be subject to neutron attenuation testing to quantify the boron carbide content for compliance with the minimum requirement in Table 1.2.2 using written procedures.

(iii) Testing of Basket

- Metamic-HT basket welds shall be tested/inspected as stated in Section 10.1.1.4 using written procedures.

Each neutron absorber plate shall be visually inspected for damage such as scratches, cracks, burrs, foreign material embedded in the surfaces, voids, and delaminations. Panels are also visually inspected for contamination on the surface as specified in the Manufacturing Manual [1.2.7]. Panels not meeting the acceptance criteria will be reworked or rejected. Unless basket is fabricated at the same factory manufacturing Metamic-HT, all panels shall be inspected before being shipped to the cask manufacturing facility where they may be subject to receipt inspection prior to installation.

FSW Procedure Qualification, Welder Operator Qualification and Welded Coupon Test:

A. Procedure qualification and welder operator qualification of the Friction Stir Welding (FSW) process shall meet the following requirements:

- The Procedure Qualification Record (PQR) shall meet the essential variable requirements of QW-267.
- The Weld Procedure Specification (WPS) shall meet the essential variable requirements of QW-267, QW-361.1(e) and QW-361.2.
- Welder operator performance qualifications shall meet the essential variable requirements of QW-361.2.
- Welder operator may be qualified by volumetric NDE of a test coupon; or a coupon from their initial production welding within the limitations of QW-304 and QW-305; or by bend tests taken from a test coupon.
- All welding by FSW process shall meet applicable requirements of ASME Section IX, 2013 Edition [8.1.1].

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- B. Procedure qualification of the Friction Stir Welding process may be accomplished by tensile testing the appropriate number of coupons per ASME Section IX (2007) and achieving a nominal 60% of Metamic-HT MGW tensile strength. Verification of weld soundness is performed by visual examination, radiography and bend testing per approved written procedures (bend testing emulates ASME Section IX). Bend test qualification of a representative weld sample emulating ASME Section IX paragraph QW 160 at a bend radius that produces at least 150% of the average tensile strain developed in the friction stir welded joint under the hypothetical free drop accident condition. The bend radius shall be recorded on the PQR. The bend test sample must meet the acceptance criteria of Section IX QW-163 and visual examination acceptance criteria of ASME Section III Subsection NG 5362 with any additional requirements per Holtec approved written procedure. In addition, at least one welded coupon from the population of Metamic-HT production panels used for manufacturing a fuel basket type must pass the criteria provided herein and shall be so documented in the Documentation Package of the manufactured fuel baskets.

#### 10.1.7 Thermal Acceptance Tests

The thermal performance of the HI-STORM FW system, including the MPCs and HI-TRAC transfer cask, is demonstrated through analysis in Chapter 4 of this FSAR. Dimensional inspections to verify the item has been fabricated to the dimensions provided in the drawings shall be performed prior to system loading.

The first manufactured MPC, either MPC-37 or MPC-89, will be thermally tested using an approved QA controlled Holtec procedure [10.1.9]. The following are the basic steps of this procedure.

1. The MPC will be arrayed in the vertical orientation on the test pad with interface insulation to minimize heat loss from the bottom.
2. Twelve storage cells (three in each quadrant) will be loaded with bayonet electric heaters each calibrated to deliver one kilowatt heat uniformly over its length. The heaters will be situated co-axially within each storage cell. Thus the heat generation in the MPC shall be quadrant-symmetric.
3. The top of the MPC shall be enclosed by an insulated lid. Calibrated thermocouples will be fastened to selected cell walls in each quadrant in a symmetric manner.
4. The test will be run for a sufficiently long time such that steady state conditions are reached. The ambient temperature and the thermocouple readings will be taken as specified in the test procedure.
5. The test condition will be simulated on the design basis FLUENT model of the MPC in Chapter 4 and the temperatures at all of the thermocouple locations predicted by FLUENT will be compared with the test data.
6. The amounts by which the FLUENT temperatures exceed the corresponding measured temperatures (positive margin) collectively define the margin of

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conservatism in the FSAR analysis model. A negative margin will warrant an immediate report to the NRC and appropriate licensing action pursuant to Holtec's QA program.

Following the loading and placement on the storage pad of the first HI-STORM system placed in service as specified in CoC Condition #8, the operability of the natural convective cooling of the HI-STORM FW system shall be verified by the performance of an air mass flow rate test. A description of the test is described in Chapter 9.

In addition, the technical specifications require periodic surveillance of the overpack air inlet and outlet vents or, optionally, implementation of an overpack air temperature monitoring program to provide continued assurance of the operability of the HI-STORM FW heat removal system.

#### 10.1.8 Cask Identification

Each MPC, HI-STORM overpack, and HI-TRAC transfer cask shall be marked with a model number, identification number (to provide traceability back to documentation), and the empty weight of the item in accordance with the marking requirements specified in 10 CFR 72.236(k).

Table 10.1.1 MPC INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	<p>a) Examination of MPC code welds per ASME Code Section III, Subsection NB, as defined on design drawings, per NB-5300, as applicable.</p> <p>b) A dimensional inspection of the internal basket assembly and canister shall be performed to verify compliance with design requirements.</p> <p>c) A dimensional inspection of the MPC lid and MPC closure ring shall be performed prior to inserting into the canister shell to verify compliance with design requirements.</p> <p>d) NDE of weldments are defined on the design drawings using standard American Welding Society NDE symbols and/or notations. Acceptance criteria for non-code welds are defined on the drawings.</p> <p>e) Cleanliness of the MPC shall be verified upon completion of fabrication.</p> <p>f) The packaging of the MPC at the completion of fabrication shall be verified prior to shipment.</p>	<p>a) The MPC shall be visually inspected prior to placement in service at the licensee's facility.</p> <p>b) MPC protection at the licensee's facility shall be verified.</p> <p>c) MPC cleanliness and exclusion of foreign material shall be verified prior to placing in the spent fuel pool.</p>	a) None.

Table 10.1.1 (continued) MPC INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Structural	a) Assembly and welding of MPC components is performed per ASME Code Section IX and III, Subsection NB, as applicable.  b) Materials analysis (steel, neutron absorber, etc.), is performed and records are kept in a manner commensurate with "important to safety" classifications.	a) None.	a) A multi-layer liquid penetrant (PT) examination of the MPC lid-to-shell weld is performed per ASME Section V, Article 2. Acceptance criteria for the examination are defined in Subsection 10.1.1, and in the Licensing Drawings.  b) ASME Code NB-6000 pressure test is performed after MPC closure welding. Acceptance criteria are defined in the Code.
Leak Tests	a) Helium leakage testing of the MPC base metal (shell, baseplate and MPC lid), MPC shell to baseplate welds and MPC shell to shell welds is performed on the unloaded MPC. Acceptance criterion is in accordance with "leaktight" definition in ANSI N14.5.	a) None.	a) Helium leakage testing is performed on the vent and drain port cover plates to MPC lid field welds. See Technical Specification for guidance on acceptance criteria.
Criticality Safety	a) The boron content is verified at the time of neutron absorber material manufacture.  b) The installation of MPC cell panels is verified by inspection.	None.	None.
Shielding Integrity	a) Material compliance is verified through CMTRs.  b) Dimensional verification of MPC lid thickness is performed.	None.	None.

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Table 10.1.1 (continued) MPC INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Thermal Acceptance	a) None.	a) None.	a) None.
Fit-Up Tests	a) Fit-up of the following components is verified during fabrication. <ul style="list-style-type: none"> <li>- MPC lid</li> <li>- vent/drain port cover plates</li> <li>- MPC closure ring</li> </ul> b) A gauge test of all basket fuel compartments.	a) Fit-up of the following components is verified during pre-operation. <ul style="list-style-type: none"> <li>-MPC lid</li> <li>-MPC closure ring</li> <li>-vent/drain cover plates</li> </ul>	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 10.1.2 HI-STORM FW OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	<p>Structural Steel Components:</p> <p>a) All structural welds shall be visually examined per ASME Section V, Article 9 with acceptance criteria per ASME Section III, Subsection NF, NF-5360.</p> <p>b) All structural welds requiring PT examination as shown on the Licensing Drawings shall be PT examined per ASME Section V, Article 6 with acceptance criteria per ASME Section III, Subsection NF, NF-5350.</p> <p>c) All structural welds requiring MT examination as shown on the drawings shall be MT examined per ASME Section V, Article 7 with acceptance criteria per ASME Section III, Subsection NF, NF-5340.</p> <p>d) NDE of weldments shall be defined on design drawings using ANSI NDE symbols and/or notations.</p> <p>Concrete Components: The following processes related to concrete components shall be implemented in accordance with the provisions of Appendix 1.D of [10.1.6]. Concrete testing shall be in accordance with Table 1.D.1. Activities shall be conducted in accordance with written and approved procedures.</p> <p>a) Assembly and examination. b) Mixing, pouring, and testing.</p>	<p>a) The overpack shall be visually inspected prior to placement in service.</p> <p>b) Fit-up with mating components (e.g., lid) shall be performed directly whenever practical or using templates or other means.</p> <p>c) overpack protection at the licensee's facility shall be verified.</p> <p>d) Exclusion of foreign material shall be verified prior to placing the overpack in service at the licensee's facility.</p>	<p>a) Indications identified during visual inspection shall be corrected, reconciled, or otherwise dispositioned.</p> <p>b) Exposed surfaces shall be monitored for coating deterioration and repair/recoat as necessary.</p>

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Table 10.1.2 (continued) HI-STORM FW OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE) (continued)	General: a) Cleanliness of the overpack shall be verified upon completion of fabrication. b) Packaging of the overpack at the completion of shop fabrication shall be verified prior to shipment.		
Structural	a) 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 category.  b) Concrete compressive strength tests shall be performed per Appendix 1.D of [10.1.6].	a) No structural or pressure tests are required for the overpack during pre-operation.	a) No structural or pressure tests are required for the overpack during operation.
Leak Tests	a) None.	a) None.	a) None.
Criticality Safety	a) No neutron absorber tests of the overpack are required for criticality safety during fabrication.	a) None.	a) None.
Shielding Integrity	a) Concrete density shall be verified per Appendix 1.D of [10.1.6], at time of placement.  b) Shell thicknesses and dimensions between inner and outer shells shall be verified as conforming to design drawings prior to concrete placement.	a) None	a) A shielding effectiveness test shall be performed after the initial fuel loading.

Table 10.1.2 (continued) HI-STORM FW OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Thermal Acceptance	a) Inner shell I.D. and vent size, configuration and placement shall be verified.	a) No pre-operational testing related to the thermal characteristics of the overpack is required.	a) Air temperature rise test(s) shall be performed after initial loading of the first HI-STORM FW system in accordance with the operating procedures in Chapter 9.  b) Periodic surveillance shall be performed by either (1) or (2) below, at the licensee's option.  (1) Inspection of overpack inlet and outlet air vent openings for debris and other obstructions. (2) Temperature monitoring.
Cask Identification	a) Verification that the overpack identification is present in accordance with the drawings shall be performed upon completion of assembly.	a) The overpack identification shall be checked prior to loading.	a) The overpack identification shall be periodically inspected per licensee procedures and repaired or replaced if damaged.
Fit-up Tests	a) Lid fit-up with the overpack shall be verified following fabrication.	a) None.	a) None.

Table 10.1.3 HI-TRAC VW TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	<p>a) All structural welds shall be visually examined per ASME Section V, Article 9 with acceptance criteria per ASME Section III, Subsection NF, NF-5360.</p> <p>b) All structural welds requiring PT examination as shown on the Design Drawings shall be PT examined per ASME Section V, Article 6 with acceptance criteria per ASME Section III, Subsection NF, NF-5350.</p> <p>c) All structural welds requiring MT examination as shown on the Design Drawings shall be MT examined per ASME Section V, Article 7 with acceptance criteria per ASME Section III, Subsection NF, NF-5340.</p> <p>d) NDE of weldments shall be defined on design drawings using standard ANSI NDE symbols and/or notations</p> <p>e) Cleanliness of the transfer cask shall be verified upon completion of fabrication.</p> <p>f) Packaging of the transfer cask at the completion of fabrication shall be verified prior to shipment.</p>	<p>a) The transfer cask shall be visually inspected prior to placement in service.</p> <p>b) Transfer cask protection at the licensee's facility shall be verified.</p> <p>c) Transfer cask cleanliness and exclusion of foreign material shall be verified prior to use.</p>	<p>a) Visual inspections of the transfer cask shall be performed to assure continued compliance with drawing requirements.</p>

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Table 10.1.3 (continued) HI-TRAC VW TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	Fabrication	Pre-operation	Maintenance and Operations
Structural	a) 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 category.  b) A pressure test of the neutron shield water jacket shall be performed upon completion of fabrication.	a) None.	None.
Leak Tests	a) None.	a) None.	a) None.
Criticality Safety	a) None.	a) None.	a) None.
Thermal Acceptance	a) The thermal properties of the transfer cask are established by calculation and inspection, and are not tested during fabrication.	a) None.	a) None
Cask Identification	a) Verification that the transfer cask identification is present in accordance with the drawings shall be performed upon completion of assembly.	a) The transfer cask identification shall be checked prior to loading.	a) The transfer cask identification shall be periodically inspected per licensee procedures and repaired or replaced if damaged.
Fit-up Tests	a) Fit-up tests of the transfer cask bottom lid shall be performed during fabrication.	None.	a) Fit-up of the bottom lid shall be verified prior to use.

Table 10.1.4 HI-STORM FW MPC NDE REQUIREMENTS			
Weld Location	NDE Requirement	Applicable Code	Acceptance Criteria (Applicable Code)
Shell longitudinal seam	RT	ASME Section V, Article 2 (RT)	RT: ASME Section III, Subsection NB, Article NB-5320
	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Shell circumferential seam	RT	ASME Section V, Article 2 (RT)	RT: ASME Section III, Subsection NB, Article NB-5320
	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Baseplate-to-shell	RT	ASME Section V, Article 2 (RT)	RT: ASME Section III, Subsection NB, Article NB-5320
	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Lid-to-shell	PT (root and final pass) and multi-layer PT.	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
	PT (surface following pressure test)		
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 NB, Article NB-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 10.1.5

**REFERENCE ASME CODES FOR CODE WELD INSPECTIONS AND INSPECTION  
CRITERIA OF HI-STORM FW COMPONENTS**

<b>Component</b>	<b>Applicable Reference Code for Inspection Criteria</b>	<b>Applicable Code for Inspection Process</b>
MPC Confinement Boundary	ASME Section III Subsection NB	Section V
HI-STORM FW Overpack Steel Weldment	ASME Section III Subsection NF for Class 3 Structures	Section V
HI-TRAC VW Transfer Cask (Steel Weldment)	ASME Section III Subsection NF for Class 3 Structures	Section V



Table 10.1.6 Metamic-HT Testing Requirements					
	Item Tested	Property Tested For	Frequency of Test	Purpose of Test	Acceptance Criterion
i.	B <sub>4</sub> C powder (raw material) (see note 1)	Particle size distribution	One sample per lot	To verify material supplier's data sheet	Per Holtec's Purchasing Specification [1.2.7]
		Purity	One sample per lot	To verify material supplier's data sheet	ASTM C-750
ii.	Al Powder (raw material)	Particle Size Distribution	One sample per lot	To verify material supplier's data sheet	Per Holtec's Purchasing Specification [1.2.8]
		Purity	One sample per lot	To verify material supplier's data sheet	Must be 99% (min.) pure aluminum
iii.	B <sub>4</sub> C/Al Mix	B <sub>4</sub> C Content (by the wet chemistry method)	One sample per mixed/blended powders lot	To ensure wt.% B <sub>4</sub> C requirements compliance	The weight density of B <sub>4</sub> C must meet the minimum wt% specification in Table 1.2.2.
iv.	Finished Metamic-HT panel	Thickness and width, straightness, camber and bow	Each Panel	To ensure fabricability of the basket	Per Holtec's Purchasing Specification [1.2.8]
		Mechanical Properties, See Table 10.1.8.	Per Sampling Plan Table 10.1.7 (see Note 2)	To ensure structural performance.	MGV per Table 1.2.8a
		B-10 areal density (by neutron attenuation)	One coupon from each Metamic-HT manufactured lot	To ensure criticality safety	The B <sub>4</sub> C content must meet the minimum wt% specification in Table 1.2.2.

## Notes:

- The B<sub>4</sub>C testing requirements apply if the raw material supplier is not in Holtec's (Or Nanotec's) Approved Vendor List.
- Sampling Plan is included in the Metamic-HT Manufacturing Manual [1.2.7].

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Table 10.1.7 Tier System for Coupon Testing		
Tier No.	Number of Extrusions Tested as a Percent of Number of Extrusions in the Lot	Number of Continuous Lots that Must Pass to Drop Down to the Next Tier
1	20	5
2	12.5	5
3	5	10
4	1	N/A
<p>Note 1: If a coupon fails with respect to any MGv property, then it may be replaced by two coupons from the extrusion that produced the failed coupon. If both of the replacement coupons pass the failed MGv property, then the lot can be accepted. If either of the replacement coupons is unsuccessful in meeting the failed MGv property, then the entire lot is rejected. As an alternative to rejecting the entire lot, testing of the failed MGv value on all extrusions within the lot is permitted to isolate acceptable panels.</p> <p>Note 2: Testing shall be moved up to the next tier if any MGv property fails in two consecutive lots.</p>		

Table 10.1.8 Minimum Guaranteed Values Required for Certification of Production Runs of Metamic-HT (All testing performed at ambient temperature.)		
	Property	MGV
1	Yield Strength, ksi	See Table 1.2.8a for MGV values
2	Tensile Strength, ksi	
3	Young's Modulus, ksi	
4	Area Reduction, %	

## 10.2 MAINTENANCE PROGRAM

An ongoing maintenance program shall be defined and incorporated into the HI-STORM FW system Operations and Maintenance Manual, which shall be prepared and issued prior to the first use of the system by a 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 regulations, the conditions in the Certificate of Compliance, and the design requirements and criteria contained in this FSAR.

The HI-STORM FW system is totally passive by design: There are no active components or monitoring systems required to assure the performance of its safety functions. As a result, only minimal maintenance will be required over its lifetime, and this maintenance would primarily result from the effects of weather. Typical of such maintenance would be the reapplication of corrosion inhibiting materials on accessible external surfaces. Visual inspection of the vent screens is required to ensure the air inlets and outlets are free from obstruction (or alternatively, temperature monitoring may be utilized). Such maintenance requires methods and procedures that are far less demanding than those currently in use at power plants.

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-STORM FW system is provided in Table 10.2.1.

### 10.2.1 Structural and Pressure Parts

Prior to each fuel loading, a visual examination in accordance with a written procedure shall be required of the HI-TRAC TALs and the bottom lid bolts and bolt holes. The examination shall inspect for indications of overstress such as cracks, deformation, wear marks, and missing or damaged threads. Repairs or replacement in accordance with written and approved procedures shall be required if an unacceptable condition is identified.

As described in Chapters 7 and 12 of this FSAR, there are no credible normal, off-normal, or accident events which can cause the structural failure of the MPC. Therefore, periodic structural or pressure tests on the MPCs following the initial acceptance tests are not required as part of the storage maintenance program.

### 10.2.2 Leakage Tests

There are no seals or gaskets used on the fully-welded MPC confinement system. As described in Chapters 7 and 12, there are no credible normal, off-normal, or accident events which can cause the failure of the MPC Confinement Boundary welds. Therefore, leakage tests are not required as part of the storage maintenance program.

### 10.2.3 Subsystem Maintenance

The HI-STORM FW system does not include any subsystems, which provide auxiliary cooling. Normal maintenance and calibration testing will be required on the vacuum drying, forced helium drying, helium backfill, and leakage testing systems per their O&M manuals. 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 with the HI-STORM FW require no maintenance. If the cask user chooses to use an air temperature monitoring system in lieu of visual inspection of the air inlet and outlet vents, the thermocouples and associated temperature monitoring instrumentation shall be maintained and calibrated in accordance with the user's QA program commensurate with the equipment's safety classification and designated QA category. See also Subsection 10.2.6.

### 10.2.4 Pressure Relief Devices

The pressure relief devices used on the water jackets for the HI-TRAC VW transfer cask shall be calibrated as specified in the HI-TRAC VW O&M Manual to ensure pressure relief settings are accurate prior to the cask's use.

### 10.2.5 Shielding

The gamma and neutron shielding materials in the HI-STORM FW overpack, HI-TRAC VW, and MPC are not subject to measurable degradation over time or as a result of usage.

Radiation monitoring of the ISFSI by the licensee in accordance with 10CFR72.104(c) provides ongoing evidence and confirmation of 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 water level in the HI-TRAC VW water jacket shall be verified during each loading campaign in accordance with the licensee's approved operations procedures.

The neutron absorber panels installed in the MPC baskets are not expected to degrade under normal long-term storage conditions. Therefore, no periodic verification testing of neutron poison material is required on the HI-STORM FW system.

### 10.2.6 Thermal

In order to assure that the HI-STORM FW system continues to provide effective thermal performance during storage operations, surveillance of the air vents (or alternatively, by temperature monitoring) shall be performed in accordance with written procedures.

For those licensees choosing to implement temperature monitoring as the means to verify overpack heat transfer system operability, a maintenance and calibration program shall be established in accordance with the plant-specific Quality Assurance Program, the equipment's quality category, and manufacturer's recommendations.

Table 10.2.1 HI-STORM 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, during storage operation
Overpack vent screen visual inspection for damage, holes, etc.	Monthly
HI-STORM FW Shielding Effectiveness Test	In accordance with Technical Specifications after initial fuel loading
HI-TRAC cavity visual inspection	Prior to each handling campaign
HI-TRAC TAL visual inspection	Prior to each handling campaign
HI-TRAC bottom lid bolts and bolt holes	Prior to each handling campaign
HI-TRAC pressure relief device calibration	Per the device manufacturer's recommendation.
HI-TRAC internal and external visual inspection for compliance with design drawings	Annually
HI-TRAC water jacket water level visual examination	During each handling campaign in accordance with licensee approved operations procedures
Overpack visual inspection of identification markings	Annually
Overpack Air Temperature Monitoring System	Per licensee's QA program and manufacturer's recommendations

## 10.3 REGULATORY COMPLIANCE

Chapter 10 of this FSAR has been prepared to summarize the commitments of Holtec International to design, construct, and test the HI-STORM FW system in conformance with the Codes and Standards identified in Chapter 2. Completion of the defined acceptance test program for each HI-STORM FW system will provide the assurance that the SSCs important to safety will perform their intended function without limitation. The performance of the maintenance program by the licensee for each loaded HI-STORM FW system will provide the 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 10.1 of this FSAR describes Holtec International's proposed program for pre-operational testing and initial operations of the HI-STORM FW system. Section 10.2 describes the proposed HI-STORM FW system's maintenance program.
2. Structures, systems, and components (SSCs) of the HI-STORM FW system designated as important to safety will be designed, fabricated, erected, assembled, inspected, tested, and maintained to quality standards commensurate with their safety category. The licensing drawings in Section 1.5 and Table 9.2.1 of this FSAR identify the safety importance and quality classifications of SSCs of the HI-STORM FW system and its ancillary equipment, respectively. Tables 1.2.6 and 1.2.7 present the applicable standards for their design, fabrication, and inspection of the HI-STORM FW system components.
3. Holtec International will examine and test the HI-STORM FW system to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness. Section 10.1 of this FSAR describes the MPC Confinement Boundary assembly, inspection, and testing.
4. Each cask shall bear a nameplate 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-STORM FW system are in compliance with 10CFR72 [10.0.1], and that the applicable acceptance criteria have been satisfied. The acceptance tests and maintenance program will provide reasonable assurance that the HI-STORM FW system will allow safe storage of spent fuel throughout its certified term. This can be concluded based on a review that considers the overarching regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

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## 10.4 REFERENCES

- [10.0.1] 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".
- [10.0.2] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", January 1997.
- [10.1.1] American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Sections II, III, V, IX, and XI, 2007 Edition.
- [10.1.2] American Society for Nondestructive Testing, "Personnel Qualification and Certification in Nondestructive Testing," Recommended Practice No. SNT-TC-1A, December 1992.
- [10.1.3] 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.
- [10.1.4] NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", U.S. Nuclear Regulatory Commission, Washington, D.C., July 1980.
- [10.1.5] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials Leakage Tests on Packages for Shipment", ANSI N14.5, January 1997.
- [10.1.6] "Final Safety Analysis Report for HI-STORM 100 Cask Storage System", Holtec Report No. HI-2002444 (latest revision).
- [10.1.7] "Metamic-HT Qualification Sourcebook", Holtec Report No. HI-2084122, Latest Revision (Holtec Proprietary)<sup>1</sup>
- [10.1.8] "Sampling Procedures and Tables for Inspection by Attributes", Military Standard MIL-STD-105E, (10/5/1989).
- [10.1.9] "HI-STORM FW MPC Thermal Test Procedure", Holtec Procedure HPP-5018-1, Rev. 0.

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<sup>1</sup>Supporting document submitted with the HI-STAR 180 License Application (Docket 71-9325).

# CHAPTER 11: RADIATION PROTECTION<sup>†</sup>

## 11.0 INTRODUCTION

This chapter discusses the design considerations and operational features that are incorporated in the HI-STORM FW system design to protect plant personnel and the public from exposure to radioactive contamination and ionizing radiation during canister loading, closure, transfer, and on-site dry storage. Occupational exposure estimates for typical canister loading, closure, transfer operations, and ISFSI inspections are provided. An off-site dose assessment for a typical ISFSI is also presented. Since the determination of off-site doses is necessarily site-specific, similar dose assessments shall be prepared by the licensee, as part of implementing the HI-STORM FW system in accordance with 10CFR72.212 [11.0.1]. The information provided in this chapter meets the requirements of NUREG-1536 [11.0.3].

## 11.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS-LOW-AS-REASONABLY-ACHIEVABLE (ALARA)

### 11.1.1 Policy Considerations

The HI-STORM FW has been designed in accordance with 10CFR72 [11.0.1] and maintains radiation exposures ALARA consistent with 10CFR20 [11.1.1] and the guidance provided in Regulatory Guides 8.8 [11.1.2] and 8.10 [11.1.3]. Licensees using the HI-STORM FW system will utilize and apply their existing site ALARA policies, procedures and practices for ISFSI activities to ensure that personnel exposure requirements of 10CFR20 [11.1.1] are met. Personnel performing ISFSI operations shall be trained on the operation of the HI-STORM FW system, and be familiarized with the expected dose rates around the MPC, HI-STORM overpack and HI-TRAC VW during all phases of loading, storage, and unloading operations. Chapter 13 provides dose rate limits at the HI-TRAC VW and HI-STORM overpack surfaces to ensure that the HI-STORM FW system is operated within design basis conditions and that ALARA goals will be met. Pre-job ALARA briefings will 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 [11.1.1] standards for radiation protection are met in accordance with the site's written commitments.

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<sup>†</sup> This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61[11.0.2]. However, the material content of this chapter also fulfills the requirements of NUREG 1536[11.0.3]. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1 in this SAR. Finally, all terms-of-art used in this chapter are consistent with the terminology of the Glossary.

It is noted that although Loading Pattern B for the MPC-37 allows assemblies with higher heat loads and therefore higher source terms in the outer region (Region 3) of the MPC, the guiding principle in selecting fuel loading should still be to preferentially place assemblies with higher source terms in the inner regions of the basket as far as reasonably possible.

### 11.1.2 Radiation Exposure Criteria

The radiological protection criteria that limit exposure to radioactive effluents and direct radiation from an ISFSI using the HI-STORM FW system are as follows:

1. 10CFR72.104 [11.0.1] 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 to the whole body, 75 mrem to the thyroid, and 25 mrem 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. As discussed below, the design features of the HI-STORM FW system components are configured to meeting this and other criteria cited below without undue burden to the user (discussed in Subsection 11.1.2).
2. 10CFR72.106 [11.0.1] 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, 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. The lens dose equivalent shall not exceed 15 rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem. The licensee is responsible for demonstrating site-specific compliance with this requirement.
3. 10CFR20 [11.1.1], Subparts C and D, limit occupational exposure and exposure to individual members of the public. The licensee is responsible for demonstrating site-specific compliance with this requirement.
4. Regulatory Position 2 of Regulatory Guide 8.8 [11.1.2] provides guidance regarding facility and equipment design features. This guidance has been followed in the design of the HI-STORM FW storage 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) [11.0.1]. Depending on the site-specific ISFSI design, other equivalent measures may be used. Unauthorized access is prevented once a loaded HI-STORM FW overpack is placed in an ISFSI. Due to the passive 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 storage cask and transfer cask biological shielding that minimizes personnel exposure, as described in Chapter 5 and in this chapter. Fundamental design considerations that most directly influence occupational exposures with dry storage systems in general and which have been incorporated into the HI-STORM FW 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 overpack that provides 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 (that come in contact with pool water) to reduce decontamination time;
  - minimization of potential crud traps on the handling equipment to reduce decontamination requirements;
  - capability of maintaining uncontaminated water in the MPC during welding to reduce dose rates;
  - capability of maintaining water in the transfer cask annulus space and water jacket to reduce dose rates during closure operations;
  - MPC penetrations located and configured to reduce neutron streaming paths;
  - elimination of trunnions in the HI-TRAC VW, which serve as streaming paths;
  - streaming paths in the HI-STORM FW overpack are limited to the air vent passages.

- MPC vent and drain ports with resealable caps to prevent the release of radionuclides during loading and unloading operations and facilitate draining, drying, and backfill operations;
  - use of a bottom lid, annulus seal, and Annulus Overpressure System to prevent contamination of the MPC shell outer surfaces during in-pool activities;
  - maximization of shielding around the top region of HI-TRAC VW where the most human activities occur during loading operations; and
  - low-maintenance design to reduce occupational dose during long-term storage.
- Regulatory Position 2c, regarding process instrumentation and controls, is met since there are no radioactive systems at an ISFSI.
  - Regulatory Position 2d, regarding control of airborne contaminants, is met since the HI-STORM FW storage system is designed to withstand all design basis conditions without loss of confinement function, as described in Chapter 7 of this SAR, 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 HI-TRAC VW-MPC annulus and by using a proven inflatable annulus seal design.
  - Regulatory Position 2e, regarding crud control, is not applicable to a HI-STORM FW 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 transfer cask is decontaminated prior to being removed from the plant's fuel building. The exterior surface of the HI-TRAC VW transfer cask is designed for ease of decontamination. In addition, an inflatable annulus seal 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 Enclosure Vessel. This material is resistant to the damaging effects of radiation and is well proven in the SNF cask service. Use of this material quantitatively reduces or eliminates the need to perform maintenance (or replacement) on the primary confinement system.

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### 11.1.3 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-STORM FW system include:

- totally-passive design requiring minimal maintenance and monitoring (other than security monitoring) during storage;
- remotely operated welding system, lift yoke, mating device and moisture removal systems to reduce time operators spend in the vicinity of the loaded MPC;
- use of a well-shielded base for staging the welding system;
- maintaining water in the MPC and the annulus region during MPC closure activities to reduce dose rates;
- low fuel assembly lift-over height over the HI-TRAC VW maximizes water coverage over assemblies during fuel assembly loading;
- a water-filled neutron shield jacket allows filling after removal of the HI-TRAC VW from the spent fuel pool. This maximizes the shielding on the HI-TRAC VW without exceeding the crane capacity;
- 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 HI-STORM FW overpack and HI-TRAC VW 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 annulus (receiving from transport) to assess the condition of the cladding and MPC Confinement Boundary;
- HI-STORM FW overpack temperature monitoring equipment allows remote monitoring of the vent operability surveillance;
- Use of proven ALARA measures such as 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 HI-TRAC VW transfer cask heat up and surveying of HI-TRAC VW prior to removal from the fuel handling building;

- Use of non-porous neutron absorber (Metamic-HT) to preclude waterlogging of the neutron absorber to minimize basket drying time. Specifically, Boral (a sandwich of aluminum sheets containing a mixture of boron carbide and aluminum powder which tends to hold the pool water in the porous space of the mixture extending canister drying times) is prohibited from use in HI-STORM FW MPCs);
- a sequence of short-term operations based on ALARA considerations; and
- use of mock-ups and dry run training to prepare personnel for actual work situations

#### 11.1.4 Auxiliary/Temporary Shielding

In addition to the design and operational features built into the HI-STORM FW system components, a number of ancillary shielding devices can be deployed to mitigate occupational dose. Ancillaries are developed on a site-specific basis that further reduce radiation at key work locations and/or allow for operations to be performed faster to reduce the time personnel spend in close proximity in the radiation field. Licensees are encouraged to use such ALARA-friendly ancillaries and practices.

## 11.2 RADIATION PROTECTION FEATURES IN THE SYSTEM DESIGN

The design of the HI-STORM FW components has been principally focused on maximizing ALARA during the short-term operations as well as during long-term storage. Some of the key design features engineered in the system components to minimize occupational dose and site boundary dose are summarized in Table 11.2.1. The design measures listed in Table 11.2.1 have been incorporated in the HI-STORM FW system to effectively reduce dose in fuel storage applications.



Table 11.2.1

DESIGN MEASURES IN THE HI-STORM FW SYSTEM COMPONENTS THAT MITIGATE DOSE			
	Component	Description of Design Feature	The Design Measure is Effective in Reducing the (A) Site Boundary Dose (B) Occupational Dose
1.	HI-STORM FW Overpack	Use of the steel weldment structure permits the density of concrete (set at a minimum of 150 lb/cubic feet) to be increased to as high as 200 lb/cubic feet.	A
2.	HI-STORM FW Overpack	The lid of the HI-STORM FW overpack contains the outlet ventilation ducts (Holtec Patent No. 6,064,710) in the overpack's closure lid. This eliminates the need for temporary shielding that will otherwise be needed if the ducts were located in the cask body for MPC transfer operations.	B
3.	HI-STORM FW Overpack	Use of multiple curved inlet ducts maximize radiation blockage (Holtec Patent No. 6,519,307B1).	B
4.	HI-STORM FW Overpack	Cask's vertical disposition and use of a thick lid (see drawing package in Section 1.5) and high density concrete minimizes skyshine.	B
5.	HI-TRAC VW/ MPC	The height of the MPC minimized for each site so that the height of HI-TRAC VW can be minimized and thus the maximum amount of lateral shielding in the cask can be incorporated consistent with the plant's crane capacity limits.	B
6.	HI-TRAC VW	Lifting trunnions located in the upper region of the transfer cask serve as neutron streaming paths in the space where human activity is necessary (welding, NDE, etc.). Eliminating trunnions and replacing them with TALs (see Glossary) eliminates steaming and aids ALARA during operations at the DAS.	B
7.	MPC	Use of Metamic-HT in the fuel basket reduces the weight of the fuel basket (in comparison to stainless steel). Thus additional shielding can be incorporated in the transfer cask whose total weight is limited by the plant's crane capacity.	B

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Table 11.2.1

DESIGN MEASURES IN THE HI-STORM FW SYSTEM COMPONENTS THAT MITIGATE DOSE			
	Component	Description of Design Feature	The Design Measure is Effective in Reducing the (A) Site Boundary Dose (B) Occupational Dose
8.	HI-STORM FW overpack/MPC	<p>The dose from a HI-STORM FW storage system is minimized because of the following advantages:</p> <ul style="list-style-type: none"> <li>a. Regionalized storage of fuel (cold fuel in the peripheral storage cells) possible because of the Metamic-HT fuel basket and the thermosiphon action-enabled MPC provides self-shielding. (Note that while loading hotter fuel in the inner cells is a requirement for some but not all loading configurations, it is preferred from an ALARA perspective.)</li> <li>b. Tight packing of overpacks on the ISFSI (that maximizes self-shielding) is possible because a large spacing between the modules is not necessary.</li> </ul>	A,B
9.	MPC, HI-TRAC VW	<p>The occupational dose from loading a HI-STORM FW overpack is minimized because of:</p> <ul style="list-style-type: none"> <li>a. A well-shielded HI-TRAC VW transfer cask.</li> <li>b. Regionalized fuel loading. (Note that while loading hotter fuel in the inner cells is a requirement for some but not all loading configurations, it is preferred from an ALARA perspective)</li> <li>c. A short water draining time (less than 2 hours) for the MPC.</li> <li>d. Reduced overall MPC welding time because the welding machine does not have to be removed and replaced to weld the secondary lid.</li> <li>e. Reduced time and personnel needed to install the MPC in the HI-STORM FW overpack due to vertical (gravity-aided) insertion.</li> <li>f. Reduced drying time because of use of porosity-free Metamic-HT.</li> </ul>	B

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Table 11.2.1			
DESIGN MEASURES IN THE HI-STORM FW SYSTEM COMPONENTS THAT MITIGATE DOSE			
	Component	Description of Design Feature	The Design Measure is Effective in Reducing the (A) Site Boundary Dose (B) Occupational Dose
10.	MPC	HI-STORM FW has been designed to accommodate high burnup and a maximum number of PWR or BWR fuel assemblies in each MPC to minimize the number of cask systems that must be handled and stored at the storage facility and later transported off-site.	A,B
11.	HI-STORM FW overpack	HI-STORM FW overpack structure is virtually maintenance free, especially over the years following its initial loading, because of the outer metal shell. The metal shell and its protective coating provide a high level of resistance degradation (e.g., corrosion).	A
12.	MPC	HI-STORM FW 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 credible accident conditions.	A,B
13.	HI-TRAC VW	HI-TRAC VW transfer cask utilizes a mating device (Holtec Patent No. 6,625,246) which reduces streaming paths and simplifies operations.	B
14.	HI-TRAC VW	The HI-TRAC VW cask and mating device are designed for quick alignment with HI-STORM.	B
15.	HI-STORM FW overpack	HI-STORM FW 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 (see Section 1.4).	A
16.	HI-STORM FW overpack	The HI-STORM FW overpack features narrow and tall optimized inlet duct shapes to minimize radiation streaming.	A

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Table 11.2.1

DESIGN MEASURES IN THE HI-STORM FW SYSTEM COMPONENTS THAT MITIGATE DOSE			
	Component	Description of Design Feature	The Design Measure is Effective in Reducing the (A) Site Boundary Dose (B) Occupational Dose
17.	HI-STORM FW overpack/MPC	The combination of a Metamic-HT (highly conductive) basket, a thermosiphon capable internal basket geometry, and a high profile inlet ducts enables the HI-STORM FW system to reject heat to the ambient to maintain the fuel cladding temperature below short-term limits in the scenario where the ISFSI is flooded and the floodwaters are just high enough to block off the ventilation airflow. This feature eliminates the need for human intervention to protect the fuel from damage from an adverse flood event and reduces occupational dose.	A,B
18.	HI-STORM FW overpack	The steel structure of the HI-STORM FW overpack gives it the fracture resistance properties that protect the overpack from developing streaming paths in the wake of the impact from a projectile such as a tornado missile strike or handling incident.	A,B

### 11.3 ESTIMATED ON-SITE CUMULATIVE DOSE ASSESSMENT

This section provides the estimates of the cumulative exposure to personnel performing loading, unloading and transfer operations using the HI-STORM FW system. This section uses the shielding analysis provided in Chapter 5, the operations procedures provided in Chapter 9 and the experience from the loading of many MPCs to develop a realistic estimate of the occupational dose.

The dose rates from the HI-STORM FW overpack, MPC lid, HI-TRAC VW, and HI-STAR 100 overpack are calculated to determine the dose to personnel during the fuel loading and unloading operations. No assessment is made with respect to background radiation since background radiation can vary significantly by site.

The estimated occupational dose is governed by three principal parameters, namely:

- i. The dose rate emanating from the MPC.
- ii. Average duration of human activity in the radiation elevated space.
- iii. Relative proximity of humans to the radiation source.

The dose rate accreted by the MPC depends on its contents. Regionalized storage has been made mandatory in the HI-STORM FW MPC to reduce its net radiation output. The duration of required human activity and the required human proximity, on the other hand, are dependent on the training level of the personnel, and user friendliness of ancillary equipment and the quality of fit-up of parts that need to be assembled in the radiation field.

To provide a uniform basis for the dose estimates presented in this chapter, the reference MPC contents data, available HI-TRAC VW weight, etc., are set down in Table 11.3.1.

Using Table 11.3.1 data, the dose data for fuel loading (wet to dry storage) is provided in Table 11.3.2. The dose for the reverse operation (dry to wet storage) is summarized in Table 11.3.3.

For each step in Table 11.3.2, the task description, average number of personnel in direct radiation field, exposure duration in direct radiation field and average dose rate are identified. The relative locations refer to all HI-STORM FW overpacks. The dose rate location points around the transfer cask and overpack were selected based on actual experience in loading HI-STORM 100 Overpacks. Cask operators typically work with workers entering and exiting the immediate cask area. To account for this, an average number of workers and average dose rates are used. The tasks involved in each step presented in Table 11.3.2 are not provided in any specific order.

### 11.3.1 Estimated Exposures for Loading and Unloading Operations

Exposures estimates presented in Tables 11.3.2 is expected to bound those for unloading operations. This assessment is based on the similarity of many loading versus operations with the elimination of several of the more dose intensive operations (such as weld inspections and leakage testing). Therefore, loading estimates should be viewed as bounding values for the contents considered for unloading operations.

### 11.3.2 Estimated Exposures for Surveillance and Maintenance

Table 11.3.3 provides an estimate of the occupational exposure required for security surveillance and maintenance of an ISFSI. Security surveillance time is based on a daily security patrol around the perimeter of the ISFSI security fence. Users may opt to utilize electronic temperature monitoring of the HI-STORM FW modules or remote viewing methods instead of performing direct visual observation of the modules. The security surveillances can be performed from outside the ISFSI, and the ISFSI fence is typically positioned such that the area outside the fence is not a radiation area. Although the HI-STORM FW system requires only minimal maintenance during storage (e.g., touch-up paint), maintenance will be required around the ISFSI for items such as security equipment maintenance, grass cutting, snow removal, vent system surveillance, drainage system maintenance, and lighting, telephone, and intercom repair, hence most of the maintenance is expected to occur outside the actual cask array.

<b>Table 11.3.1</b> <b>ASSUMED PARAMETERS FOR DOSE ESTIMATE UNDER SHORT-TERM OPERATIONS AND UNDER LONG-TEM STORAGE</b>		
	<b>Item</b>	<b>Value</b>
<b>1.</b>	<b>MPC-Contents (MPC-37)<sup>1</sup></b>	<b>45,000 MWD/MTU and 4.5 years</b>
<b>2.</b>	<b>Weight of HI-TRAC VW Full of Fuel and Water</b>	<b>125 tons</b>
<b>3.</b>	<b>HI-STORM Concrete Density</b>	<b>150 lb/cubic feet</b>

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<sup>1</sup> The case of MPC-37 is used but similar results are expected for the MPC-89.

<b>TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM</b>				
<b>Task Description (See Chapter 9 for detailed description of operations)</b>	<b>Average Number of Personnel in Direct Radiation Field</b>	<b>Exposure Duration in Direct Radiation Field (mins)</b>	<b>Average Dose Rate at worker location (mrem/hr)</b>	<b>Exposure (mrem)</b>
Fuel loading and removal of the transfer cask and MPC from the spent fuel pool (includes: fuel loading, fuel assembly identification check, MPC lid installation, Lift Yoke attachment to the HI-TRAC VW, HI-TRAC VW removal from the spent fuel pool, preliminary decontamination, HI-TRAC VW movement to the DAS, Lift Yoke removal and decontamination. Background radiation of 1 mrem/hr assumed.	3	800	1.0	40.0
MPC preparation for closure (includes: HI-TRAC VW and MPC decontamination, radiation surveys, partial MPC pump down, annulus seal removal, partial lowering of annulus water level, annulus shield ring installation, weld system installation); workers assumed to be on scaffolding near the top of the HI-TRAC.	3	30	55.7	83.5
MPC Closure (includes MPC lid to shell welding, weld inspection). Assumes welding machine uses standard Holtec pedestal which provides additional shielding. Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 10% of the total duration.	2	185	55.7	34.3

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**TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM**

<b>Task Description (See Chapter 9 for detailed description of operations)</b>	<b>Average Number of Personnel in Direct Radiation Field</b>	<b>Exposure Duration in Direct Radiation Field (mins)</b>	<b>Average Dose Rate at worker location (mrem/hr)</b>	<b>Exposure (mrem)</b>
MPC Preparation for Storage (includes: MPC hydrostatic testing, draining, drying and backfill, vent and drain port cover plate installation, welding, weld inspection and leakage testing). Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 20% of the total duration.	2	170	175.4	198.7
MPC Closure Ring Installation (includes: closure ring to MPC shell welding, weld inspection and leakage testing of the MPC primary closure). Holtec auxiliary shielding methods and equipment assumed (lead blankets, water shields, etc.) Assumes operators are present for 10% of the total duration.	2	80	229.4	61.2
HI-STORM FW system preparation for receiving MPC (includes: HI-STORM FW overpack positioning at transfer location, HI-STORM lid removal, Mating Device installation on HI-STORM FW overpack).	3	160	0	0

<b>TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM</b>				
<b>Task Description (See Chapter 9 for detailed description of operations)</b>	<b>Average Number of Personnel in Direct Radiation Field</b>	<b>Exposure Duration in Direct Radiation Field (mins)</b>	<b>Average Dose Rate at worker location (mrem/hr)</b>	<b>Exposure (mrem)</b>
MPC Transfer (attachment of MPC lifting device, movement of HI-TRAC VW to transfer location, placement of HI-TRAC VW in Mating Device, bottom lid removal, MPC lowering, HI-TRAC VW removal, MPC lift device removal). Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 10% of the total duration.	3	120	148	88.8
HI-STORM FW overpack movement to the ISFSI (will include: movement of the HI-STORM FW overpack from the fuel building to placement of the HI-STORM FW overpack on the ISFSI pad, disconnecting transporter, attachment of HI-STORM FW lid, attachment of thermal monitoring system). Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 50% of the total duration.	3	220	37.3	205.2
<b>TOTAL EXPOSURE (person-mrem)</b>	<b>711.6</b>			

Table 11.3.3				
ESTIMATED EXPOSURES FOR HI-STORM FW SURVEILLANCE AND MAINTENANCE				
ACTIVITY	ESTIMATED PERSONNEL	ESTIMATED HOURS PER YEAR	ESTIMATED DOSE RATE (MREM/HR)	OCCUPATIONAL DOSE TO INDIVIDUAL (PERSON-MREM)
SECURITY SURVEILLANCE	1	30	3	90
ANNUAL MAINTENANCE	2	15	10	300

Notes for Tables 11.3.2, 11.3.3, AND 11.3.4:

1. Refer to Chapter 9 for detailed description of activities.
2. Number of operators may be set to 1 to simplify calculations where the duration is indirectly proportional to the number of operators. The total dose is equivalent in both respects.

## 11.4 ESTIMATED CONTROLLED AREA BOUNDARY DOSE ASSESSMENT

### 11.4.1 Controlled Area Boundary Dose for Normal Operations

10CFR72.104 [11.0.1] limits the annual dose equivalent to any real individual at the controlled area boundary to a maximum of 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem 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-STORM FW 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 [11.0.1]. 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 5.1.3 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 [11.0.1] specifies that the minimum distance from the ISFSI to the controlled area boundary is 100 meters. Therefore this is the minimum distance analyzed in Chapter 5. One hundred percent (100%) occupancy (8760 hours) is conservatively assumed. In the calculation of the annual dose, the casks were positioned on an infinite slab of soil to account for earth-shine effects. Table 5.1.3 and Figure 5.1.3 in Chapter 5 show the annual dose rates for an array of HI-STORM FWs with MPC-37. These results are presented only as an illustration to demonstrate that the HI-STORM FW system is in compliance with 10CFR72.104 [11.0.1]. 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 [11.0.1] contributors and 10CFR20 [11.1.1].

Chapter 7 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 has been rendered 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. This evaluation will account for the location of the controlled area boundary, the total number of casks on the ISFSI and the effects of the radiation from uranium fuel cycle operations within the region.

### 11.4.2 Controlled Area Boundary Dose for Off-Normal Conditions

As demonstrated in Chapter 12, the postulated off-normal conditions (off-normal pressure, off-normal environmental temperatures, leakage of one MPC weld, partial blockage of air inlets, and

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off-normal handling of HI-TRAC VW) do not result in the degradation of the HI-STORM FW system shielding effectiveness. Therefore, the dose at the controlled area boundary from direct radiation for off-normal conditions is equal to that of normal conditions.

#### 11.4.3 Controlled Area Boundary Dose for Accident Conditions

10CFR72.106 [11.0.1] specifies the maximum doses allowed to any individual at the controlled area boundary from any design basis accident (See Subsection 11.1.2). In addition, it is specified that the minimum distance from the ISFSI to the controlled area boundary be at least 100 meters.

Chapter 12 presents the results of the evaluations performed to demonstrate that the HI-STORM FW system can withstand the effects of all accident conditions and natural phenomena without the corresponding radiation doses exceeding the requirements of 10CFR72.106 [11.0.1]. The accident events addressed in Chapter 12 include: handling accidents, tip-over, fire, tornado, flood, earthquake, 100 percent fuel rod rupture, Confinement Boundary leakage, explosion, lightning, burial under debris, extreme environmental temperature, and blockage of MPC basket air inlets.

The worst-case shielding consequence of the accidents evaluated in Chapter 12 for the loaded HI-STORM FW overpack assumes that as a result of a fire, the outer-most one inch of the concrete experiences temperatures above the concrete's design temperature. Therefore, the shielding effectiveness of this outer-most one inch of concrete is degraded. However, with the available concrete providing shielding, the loss of one inch will have a negligible effect on the dose at the controlled area boundary.

The worst case shielding consequence of the accidents evaluated in Chapter 12 for the loaded HI-TRAC VW transfer cask assumes that as a result of a fire, tornado missile, or handling accident, that all the water in the water jacket is lost. The shielding analysis of the HI-TRAC VW with complete loss of the water from the water jacket is discussed in Subsection 5.1.1. The results in that subsection show the resultant dose rate at the 100-meter controlled area boundary during the accident condition. At the calculated dose rate, Table 5.1.9 shows the calculated time to reach 5 rem. This length of time is sufficient to implement and complete the corrective actions outlined in Chapter 12. Therefore, the dose requirement of 10CFR72.106 [11.0.1] is satisfied. Users will need to perform site-specific analysis considering the actual site boundary distance and fuel characteristics.

Table 11.4.1  
(Intentionally Deleted)

## 11.5 REFERENCES

- [11.0.1] *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."
- [11.0.2] Regulatory Guide 3.61 (Task CE306-4) "Standard Format for a Topical Safety Analysis Report for a Spent Fuel Storage Cask", USNRC, February 1989
- [11.0.3] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", U.S. Nuclear Regulatory Commission, January 1997.
- [11.1.1] *U.S. Code of Federal Regulations*, Title 10, "Energy" Part 20 "Standards for Protection Against Radiation."
- [11.1.2] 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.
- [11.1.3] 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 12<sup>†</sup>: ACCIDENT ANALYSIS

### 12.0 INTRODUCTION

This chapter presents the evaluation of the HI-STORM FW System for the effects of off-normal and postulated accident conditions; and other scenarios that warrant safety analysis (such as MPC reflood during fuel unloading operations), pursuant to the guidelines in NUREG-1536. The design basis off-normal and postulated accident events, including those based on non-mechanistic postulation as well as those caused by natural phenomena, are identified. For each postulated event, the event cause, means of detection, consequences, and corrective actions are discussed and evaluated. For other miscellaneous events (i.e., those not categorized as either design basis off-normal or accident condition events), a similar outline for safety analysis is followed. As applicable, the evaluation of consequences includes the impact on the structural, thermal, shielding, criticality, confinement, and radiation protection performance of the HI-STORM FW System due to each postulated event.

The structural, thermal, shielding, criticality, and confinement features and performance of the HI-STORM FW System under the short-term operations and various conditions of storage are discussed in Chapters 3, 4, 5, 6, and 7. The evaluations provided in this chapter are based on the design features and analyses reported therein.

Chapter 12 is in full compliance with NUREG-1536; no exceptions are taken.

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<sup>†</sup> 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. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the Glossary.

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## 12.1 OFF-NORMAL CONDITIONS

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 off-normal conditions are described in Subsection 2.2.2.

The following off-normal operation events have been considered in the design of the HI-STORM FW:

1. Off-Normal Pressure
2. Off-Normal Environmental Temperatures
3. Leakage of One Seal
4. Partial Blockage of Air Inlets
5. Malfunction of FHD System

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-STORM FW System can withstand the effects of off-normal events and remain in compliance with the applicable acceptance criteria. The following subsections present the evaluation of the HI-STORM FW System for the design basis off-normal conditions that demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses meet the requirements of 10CFR72.104(a) and 10CFR20, with appropriate margins.

### 12.1.1 Off-Normal Pressure

The sole pressure boundary in the HI-STORM FW System is the MPC enclosure vessel. The off-normal pressure condition is specified in Subsection 2.2.2. The off-normal pressure for the MPC internal cavity is a function of the initial helium fill pressure and the temperature reached within the MPC cavity under normal storage. The MPC internal pressure under the off-normal condition is evaluated with 10% of the fuel rods ruptured and with 100% of ruptured rods fill gas and 30% of ruptured rods fission gases released to the cavity.

#### 12.1.1.1 Postulated Cause of Off-Normal Pressure

After fuel assembly loading, the MPC is drained, dried, and backfilled with an inert gas (helium) to assure long-term fuel cladding integrity during dry storage. Therefore, the probability of failure of intact fuel rods in dry storage is extremely low. Nonetheless, the event is postulated and evaluated.

#### 12.1.1.2 Detection of Off-Normal Pressure

The HI-STORM FW System is designed to withstand the MPC off-normal internal pressure without any effects on its ability to meet its safety requirements. There is no requirement or safety imperative for detection of off-normal pressure and, therefore, no monitoring is required.

#### 12.1.1.3 Analysis of Effects and Consequences of Off-Normal Pressure

The MPC off-normal internal pressure is reported in Subsection 4.6.1 for the following conditions: limiting fuel storage scenario, tech. spec. maximum helium backfill pressure with a 10% rod rupture that causes a 100% of the ruptured rod fill gas and 30% of the ruptured rod gaseous fission products released into the MPC cavity along with off-normal ambient temperature. The analysis shows that the MPC pressure remains below the design MPC internal pressure (given in Table 2.2.1). The corresponding fuel cladding temperature is provided in Table 4.6.1. It should be noted that this bounding temperature rise does not take any credit for the increase in thermosiphon action that would accompany the pressure increase that results from both the temperature rise and the addition of the gaseous fission products to the MPC cavity. As any such increase in thermosiphon action would reduce the temperature rise, therefore the calculated pressure is higher than that would actually occur.

##### i. Structural

The structural evaluation of the MPC enclosure vessel for off-normal internal pressure conditions is discussed in Section 3.4. The stresses resulting from the off-normal pressure are confirmed to be bounded by the applicable pressure boundary stress limits.

##### ii. Thermal

The MPC internal pressure for off-normal conditions is reported in Subsection 4.6.1. The design basis internal pressure used in the structural evaluation (Table 2.2.1) bounds the off-normal condition pressure.

##### iii. Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

##### iv. Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

v. 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 pressure boundary stresses remain within allowable ASME Code values, assuring Confinement Boundary integrity.

vi. 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-STORM FW System.

12.1.1.4 Corrective Action for Off-Normal Pressure

The HI-STORM FW System is designed to withstand the off-normal pressure without any effects on its ability to maintain safe storage conditions. Therefore, there is no corrective action requirement for off-normal pressure.

12.1.1.5 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.

12.1.2 Off-Normal Environmental Temperatures

The HI-STORM FW System is designed for use at any site in the United States. Off-normal environmental temperatures of -40 to 100°F (loaded HI-STORM FW overpack) and 0°F to 100°F (loaded HI-TRAC VW transfer cask) have been conservatively selected to bound off-normal temperatures at these sites. The off-normal temperature range affects the entire HI-STORM FW System and must be evaluated against the allowable component design temperatures. The off-normal temperatures are evaluated against the off-normal condition temperature limits for HI-STORM FW components listed in Table 2.2.3.

12.1.2.1 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-STORM FW System to achieve thermal equilibrium. Because of the large mass of the HI-STORM FW System with its corresponding large thermal inertia and the limited duration for the off-normal temperatures, this assumption is conservative.

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### 12.1.2.2 Detection of Off-Normal Environmental Temperatures

The HI-STORM FW 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 for the HI-STORM FW overpack and MPC. Chapter 2 provides operational limitations on the use of the HI-TRAC VW transfer cask at temperatures  $\leq 32^{\circ}\text{F}$  and prohibits use of the HI-TRAC VW transfer cask below  $0^{\circ}\text{F}$ .

### 12.1.2.3 Analysis of Effects and Consequences of Off-Normal Environmental Temperatures

The off-normal event considers an environmental temperature of  $100^{\circ}\text{F}$  with insolation for sufficient duration to reach thermal equilibrium. The evaluation is performed for a limiting fuel storage configuration. The Off-Normal ambient temperature condition is evaluated in Subsection 4.6.1. The results are in compliance with off-normal pressure and temperature limits in Tables 2.2.1 and 2.2.3, respectively.

The off-normal event considering an environmental temperature of  $-40^{\circ}\text{F}$  and no solar insolation for a sufficient duration to reach thermal equilibrium is evaluated with respect to material design temperatures of the HI-STORM FW overpack. The HI-STORM FW overpack and MPC are conservatively assumed to reach  $-40^{\circ}\text{F}$  throughout the structure. The minimum off-normal environmental temperature specified for the HI-TRAC VW transfer cask is  $0^{\circ}\text{F}$  and the HI-TRAC VW is conservatively assumed to reach  $0^{\circ}\text{F}$  throughout the structure. Subsection 3.1.2, details the structural analysis and testing performed to assure prevention of brittle fracture failure of the HI-STORM FW System.

#### i. Structural

The effect on the MPC for the upper off-normal thermal conditions (i.e.,  $100^{\circ}\text{F}$ ) is an increase in the internal pressure. As shown in Subsection 4.6.1, the resultant pressure is below the off-normal design pressure (Table 2.2.1). The stresses resulting from the off-normal pressure are confirmed to be bounded by the applicable pressure boundary stress limits. The effect of the lower off-normal thermal conditions (i.e.,  $-40^{\circ}\text{F}$ ) requires an evaluation of the potential for brittle fracture. Such an evaluation is presented in Subsection 3.1.2.

#### ii. Thermal

The resulting off-normal system and fuel assembly cladding temperatures for the hot conditions are provided in Subsection 4.6.1 for the HI-STORM FW overpack and MPC. The evaluation in Subsection 4.6.1 indicates that all temperatures for the off-normal environmental temperatures event are within the allowable values for off-normal conditions listed in Table 2.2.3.

### iii. Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

### iv. Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

### v. 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 pressure boundary stresses remain within allowable ASME Code values, assuring Confinement Boundary integrity.

### vi. 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-STORM FW System.

#### 12.1.2.4 Corrective Action for Off-Normal Environmental Temperatures

The HI-STORM FW System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. For ambient temperatures from 0° to 32°F, ethylene glycol fortified water must be used in the water jacket of the HI-TRAC VW transfer cask to prevent freezing. There are no corrective actions required for off-normal environmental temperatures.

#### 12.1.2.5 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.

### 12.1.3 Leakage of One Seal

The HI-STORM FW System has a high integrity welded boundary to contain radioactive fission products within the Confinement Boundary. The Confinement Boundary is defined by the MPC shell, baseplate, MPC lid, vent and drain port cover plates, closure ring, and associated welds. The

closure ring provides a redundant welded closure to further protect against the release of radioactive material from the MPC cavity through the field-welded MPC lid closures. Confinement boundary welds are inspected by radiography or ultrasonic examination except for field welds that are examined by the liquid penetrant method on the root (for multi-pass welds) and final pass, at a minimum. The fabrication shop welds for the confinement boundary are tested for helium leakage. Field welds are performed on the MPC lid, the MPC vent and drain port covers, and the MPC closure ring. The welds on the vent and drain port cover plates are helium leakage tested. Additionally, the MPC lid weld is subjected to a pressure test to verify its integrity. 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 Confinement Boundary is considered non-credible.

#### 12.1.4 Partial Blockage of Air Inlets

The HI-STORM FW System is designed with debris screens on the inlet and outlet air openings. These screens ensure the openings are protected from the incursion of foreign objects. There are multiple inlet openings and an axisymmetric outlet and it is highly unlikely that blowing debris during normal or off-normal operation could block all air inlet openings. As required by the design criteria presented in Chapter 2, it is conservatively assumed that 50% of the air inlet openings are completely blocked. The scenario of the partial blockage of air inlets is evaluated with a normal ambient temperature of 80°F (Table 2.2.2), full solar insolation, and maximum SNF decay heat values. This condition is analyzed to demonstrate the thermal performance of the HI-STORM FW System during this event.

##### 12.1.4.1 Postulated Cause of Partial Blockage of Air Inlets

The presence of screens prevents foreign objects from entering the openings and the screens are either inspected periodically or the outlet air temperature is monitored per the technical specifications. It is, however, possible that blowing debris may partially block the inlet openings for a short time until the openings are cleared of debris.

##### 12.1.4.2 Detection of Partial Blockage of Air Inlets

The detection of the partial blockage of air inlet openings will occur during the routine visual inspection of the screens or temperature monitoring of the outlet air required by the technical specifications. The frequency of inspection is based on an assumed complete blockage of all air inlet openings. There is no inspection requirement as a result of the postulated partial inlet blockage, because the complete blockage of all air inlet openings is bounding.

#### 12.1.4.3 Analysis of Effects and Consequences of Partial Blockage of Air Inlets

##### i. Structural

There are no structural consequences as a result of this off-normal event since the HI-STORM FW components do not exceed the off-normal temperature limits (Table 2.2.3).

##### ii. Thermal

The thermal analysis for the 50% blocked inlet openings off-normal condition is performed in Subsection 4.6.1. The analysis demonstrates that under bounding (steady-state) conditions, no system components exceed the off-normal temperature limits in Table 2.2.3.

##### iii. Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

##### iv. Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

##### v. Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

##### vi. 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 partial blockage of air inlet openings does not affect the safe operation of the HI-STORM FW System.

#### 12.1.4.4 Corrective Action for Partial Blockage of Air Inlets

The corrective action for the partial blockage of air inlet openings is the removal, cleaning, and replacement of the affected mesh screens. After clearing of the blockage, the storage module temperatures will return to the normal temperatures reported in Chapter 4. Partial blockage of air inlet openings does not affect the safe operation of the HI-STORM FW System.

Periodic inspection of the HI-STORM FW overpack air opening screens is required per the technical specifications. Alternatively, per the technical specifications, the outlet air temperature is monitored. The frequency of inspection is based on an assumed blockage of all air inlet openings analyzed in Section 12.2.

#### 12.1.4.5 Radiological Impact of Partial Blockage of Air Inlets

The off-normal event of partial blockage of the air inlet openings has no radiological impact because the confinement barrier is not breached and the system's shielding effectiveness is not diminished.

### 12.1.5 Malfunction of FHD System

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

#### 12.1.5.1 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 trips the FHD System. In both cases a stoppage of forced helium circulation occurs. Such a circulation stoppage does not result in helium leakage from the MPC or the FHD.

#### 12.1.5.2 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.

#### 12.1.5.3 Analysis of Effects and Consequences of FHD Malfunction

##### i. Structural

The FHD System is required to be equipped with safety relief devices\* to prevent the MPC structural boundary pressures from exceeding the normal condition pressure limits. Consequently there is no adverse effect.

##### ii. Thermal

Malfunction of the FHD System is categorized as an off-normal condition, for which the applicable peak cladding temperature limit (see Table 2.2.3) must not be exceeded. The FHD System malfunction event is evaluated assuming the following bounding conditions:

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\* The relief pressure is below the off-normal design pressure (Table 2.2.1) to prevent MPC overpressure and above 7 atm to enable MPC pressurization for adequate heat transfer.



- a. Steady state maximum temperatures have been reached
- b. Design maximum heat load in the limiting MPC-37
- c. Air in the HI-TRAC VW annulus
- d. The helium pressure in the MPC is at the minimum possible value from the technical specification.

The results of a steady state analysis (which implies an extended period of FHD unavailability) are provided in Section 4.6. The results provide the assurance that the peak fuel cladding temperature in the MPC will remain below the ISG-11 limit (see Table 2.2.3) in the event of a prolonged unavailability of the FHD system under the most thermally adverse conditions (highest possible heat load absence of any forced heat removal measures and minimum system helium pressure).

### iii. Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

### iv. Criticality

There is no effect on the criticality control of the system as a result of this off-normal event.

### v. 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 MPC structural boundary internal pressures cannot exceed the normal condition pressure limits, assuring Confinement Boundary integrity.

### vi. 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-STORM FW System.

#### 12.1.5.4 Corrective Action for FHD Malfunction

The HI-STORM FW System is designed to withstand the FHD malfunction without an adverse effect on its safety functions. Consequently no corrective action is required.

#### 12.1.5.5 Radiological Impact of FHD Malfunction

The event has no radiological impact because the confinement barrier and shielding integrity are not affected.

## 12.2 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-STORM FW System or events postulated because their consequences may affect the public health and safety. Subsection 2.2.3 defines the design basis accidents considered. By analyzing for these design basis events, safety margins inherently provided in the HI-STORM FW System design can be quantified.

The results of the evaluations performed herein demonstrate that the HI-STORM FW System can withstand the effects of all credible and hypothetical accident conditions and natural phenomena without affecting safety function, and are in compliance with the acceptable criteria. In the following, the evaluation of the design basis postulated accident conditions and natural phenomena is presented. The evaluations 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 load combinations evaluated for postulated accident conditions are defined in Table 2.2.13. The accident load combination evaluations are provided in Section 3.4.

Table 12.2.1 provides a listing of the accident events considered in this section and their probability of occurrence.

### 12.2.1 HI-TRAC VW Transfer Cask Handling Accident

During the operation of the HI-STORM FW System, the loaded HI-TRAC VW transfer cask is lifted and handled in a vertical orientation at all times (with the rare handling exception of the transfer cask as described in Subsection 4.5.1). A vertical drop of the loaded HI-TRAC VW transfer cask is not a credible accident as the loaded HI-TRAC VW transfer cask shall be lifted and handled in the vertical orientation by devices designed in accordance with the criteria specified in Subsection 2.3.3 to prevent uncontrolled lowering. Therefore, postulating an uncontrolled lowering of a HI-TRAC VW transfer cask in the realm of Part 72 operations is non-credible.

### 12.2.2 HI-STORM FW Overpack Handling Accident

During the operation of the HI-STORM FW System, the loaded HI-STORM FW overpack is lifted and handled in a vertical orientation at all times. A vertical drop of the loaded HI-STORM FW is not a credible accident as the loaded HI-STORM FW shall be lifted and handled in the vertical orientation by devices designed in accordance with the criteria specified in Subsection 2.3.3 to prevent uncontrolled lowering. Therefore, postulating an uncontrolled lowering of a HI-STORM FW in the Part 72 space is non-credible.

### 12.2.3 HI-STORM FW Overpack Non-Mechanistic Tip-Over

The freestanding HI-STORM FW storage overpack, containing a loaded MPC, cannot tip over as a

result of postulated natural phenomenon events, 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 is analyzed (Subsection 2.2.3) in Chapter 3.

#### 12.2.3.1 Cause of Tip-Over

The tip-over accident is stipulated as a non-mechanistic accident because a credible mechanism for the cask to tip over cannot be identified. Detailed discussions are provided in Subsections 3.1.2 and 3.4.4.

However, it is recognized that the mechanical loadings at a specific ISFSI may be sufficiently strong to cause a tip-over event, even though such a scenario is determined to be counterfactual under the loads treated in this FSAR. To enable the safety evaluation of a postulated tip-over scenario, it is necessary to set down an analysis methodology and the associated acceptance criteria. In Sections 2.2 and 3.4, the methodology and acceptance criteria are presented and a reference tip-over problem is solved. The reference tip-over problem corresponds to a free rotation of the HI-STORM FW overpack from the condition of rest at the incipient tipping point (i.e., C.G.-over-corner). The evaluations presented below refer to the above non-mechanistic tip-over scenario.

#### 12.2.3.2 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 Subsection 3.4.4. The structural analysis demonstrates the following:

- (i) The lateral plastic deformation of the basket panels in the active fuel region is less than the limiting value in Table 2.2.11.
- (ii) The impact between the MPC guide tubes and the MPC does not cause a thru-wall penetration of the MPC shell.

The side impact will cause some localized damage to the concrete and outer shell of the overpack in the local area of impact. However, there is no significant adverse effect on the structural, confinement, thermal, or criticality performance.

As mentioned earlier the non-mechanistic tip-over accident has been addressed to demonstrate the defense-in-depth features of the design.

#### 12.2.3.3 Tip-over Accident Corrective Actions

Corrective action after a tip-over would include a radiological and visual inspection to determine the extent of the damage to the overpack and the contained MPC. Special handling procedures, including the use of temporary shielding, will be developed and approved by the ISFSI operator.

## 12.2.4 Fire

### 12.2.4.1 Cause of Fire

The potential of a fire accident near an ISFSI pad is considered to be rendered extremely remote by ensuring that there are no significant combustible materials in the area. The only credible concern is related to a transport vehicle fuel tank fire engulfing the loaded HI-STORM FW overpack or HI-TRAC VW transfer cask during their handling.

### 12.2.4.2 Fire Analysis

The HI-STORM FW System must withstand elevated temperatures due to a fire event. The HI-STORM FW overpack and HI-TRAC VW transfer cask fire accidents for storage are conservatively postulated as described in Subsection 4.6.2. The acceptance criteria for the fire accident are provided in Subsection 2.2.3.

#### 12.2.4.2.1 Fire Analysis for HI-STORM FW Overpack

The analysis for the fire accident including the methodology has been provided in Subsection 4.6.2. The transport vehicle fuel tank fire has been analyzed to evaluate the outer layers of the storage overpack heated by the incident thermal radiation and forced convection heat fluxes and to evaluate fuel cladding and MPC temperatures.

#### i. Structural

As discussed in Section 3.4, there are no structural consequences as a result of the fire accident condition since the short-term temperature limit on great majority of the concrete is not exceeded and all component temperatures remain within applicable temperature limits (Table 2.2.3). The MPC structural boundary remains within normal condition internal pressure and temperature limits.

#### ii. Thermal

Based on a conservative analysis discussed in Subsection 4.6.2, of the HI-STORM FW System response to the hypothetical fire event, it is concluded that the fire event does not significantly affect the temperature of the MPC or contained fuel. Furthermore, the ability of the HI-STORM FW System to maintain cooling of the spent nuclear fuel within temperature limits (Table 2.2.3) during and after fire is not compromised.

#### iii. Shielding

With respect to concrete damage from a fire, NUREG-1536 (4.0,V,5.b) states: “the loss of a small amount of shielding material is not expected to cause a storage system to exceed the regulatory requirements in 10 CFR 72.106 and, therefore, need not be estimated or evaluated

in the FSAR.” Less than one-inch of the overpack concrete (~4% of the overpack radial concrete thickness) is computed to exceed the short-term temperature limit therefore the effect of this small amount of degraded (not lost) shielding material is not estimated or evaluated in this FSAR.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event since the structural integrity of the confinement boundary is unaffected.

vi. Radiation Protection

Since there is minimal reduction, if any, in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the overpack fire accident does not affect the safe operation of the HI-STORM FW System.

#### 12.2.4.2.2 Fire Analysis for HI-TRAC VW Transfer Cask

To demonstrate the fuel cladding and MPC pressure boundary integrity under an exposure to a hypothetical short duration fire event during on-site handling operations, a fire accident analysis of the loaded HI-TRAC VW transfer cask is performed. The analysis for the fire accident including the methodology has been provided in Subsection 4.6.2.

i. Structural

As discussed in Section 3.4, there are no adverse structural consequences as a result of the fire accident condition.

ii. Thermal

The thermal analysis of the MPC in the HI-TRAC VW transfer cask under a fire accident is performed in Subsection 4.6.2. The analysis shows that the MPC internal pressure and fuel temperature increases as a result of the fire accident. The fire accident MPC internal pressure and peak fuel cladding temperature are substantially less than the accident limits for MPC internal pressure and maximum cladding temperature (Tables 2.2.1 and 2.2.3).

As can be concluded from the analysis, the temperatures for fuel cladding and components are below the accident temperature limits.

### iii. Shielding

The conservatively assumed loss of all the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The shielding evaluation presented in Chapter 5 demonstrates that the requirements of 10CFR72.106 are not exceeded.

### iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

### v. Confinement

There is no effect on the confinement function of the MPC as a result of this event, since the internal pressure does not exceed the accident condition design pressure and the MPC Confinement Boundary temperatures do not exceed the short-term allowable temperature limits.

### vi. Radiation Protection

There is no degradation in confinement capabilities of the MPC, as discussed above. Increases in the local dose rates adjacent to the water jacket are evaluated in Chapter 5. Immediately after the fire accident a radiological inspection of the HI-TRAC VW transfer cask shall be performed and temporary shielding shall be installed if necessary to limit exposure to site personnel.

## 12.2.4.3 Fire Dose Calculations

The complete loss of the HI-TRAC VW transfer cask neutron shield along with the water jacket shell is assumed in the shielding analysis for the post-accident analysis of the loaded HI-TRAC VW transfer cask in Chapter 5 and bounds the determined fire accident consequences. The loaded HI-TRAC VW transfer cask following a fire accident meets the accident dose rate requirement of 10CFR72.106.

The elevated temperatures experienced by the HI-STORM FW overpack concrete shield are limited to the outermost layer of steel and concrete. Therefore, overall reduction in neutron shielding capabilities is quite small. Any increase in the neutron dose rate as a result of the concrete in the outer inch reaching elevated temperatures will not significantly increase the site boundary dose rate. This is due to the limited amount of the concrete shielding which is affected and the already low site boundary dose rates. The loaded HI-STORM FW overpack following a fire accident meets the accident dose rate requirement of 10CFR72.106.

The analysis of the fire accident shows that the MPC Confinement Boundary is not compromised and therefore, there is no release of airborne radioactive materials.

#### 12.2.4.4 Fire Accident Corrective Actions

Upon detection of a fire adjacent to a loaded HI-TRAC VW transfer cask or HI-STORM FW overpack, the ISFSI owner shall take the appropriate immediate actions necessary to extinguish the fire. Fire fighting personnel should take appropriate radiological precautions, particularly with the HI-TRAC VW transfer cask as the water jacket rupture discs may open with resulting water loss and increase in radiation doses. Following the termination of the fire, a visual and radiological inspection of the equipment shall be performed.

As appropriate, temporary shielding around the HI-TRAC VW transfer cask shall be installed. Specific attention shall be taken during the inspection of the water jacket of the HI-TRAC VW transfer cask. If damage to the HI-TRAC VW transfer cask is limited to the loss of water in the water jacket due to the pressure increase, the water may be replaced. If damage to the HI-TRAC VW transfer cask is extensive and/or radiological conditions require (based on dose rate measurements), the HI-TRAC VW transfer cask shall be unloaded in accordance with Chapter 9, prior to repair.

If damage to the HI-STORM FW storage overpack as the result of a fire event is widespread and/or as radiological conditions require (based on dose rate measurements), the MPC shall be removed from the HI-STORM FW overpack in accordance with Chapter 9. However, the thermal analysis described herein demonstrates that only the outermost layer of the radial concrete which is behind the carbon steel outer shell exceeds its design temperature. The HI-STORM FW overpack may be returned to service after appropriate restoration (reapplication of coatings etc.) if there is no significant increase in the measured dose rates (i.e., the shielding effectiveness of the overpack is confirmed) and if the visual inspection is satisfactory.

#### 12.2.5 Partial Blockage of MPC Basket Flow Holes

Each MPC basket fuel cell wall has flow holes near the bottom to allow thermosiphon action to assist the cooling of MPC internals. 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 HI-STORM FW MPC basket flow holes is not credible.

#### 12.2.6 Tornado

##### 12.2.6.1 Cause of Tornado

The HI-STORM FW System will be stored on an unsheltered ISFSI concrete pad and thus will be

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subject to ambient environmental conditions throughout the storage period. Additionally, the transfer of the MPC between the HI-TRAC VW transfer cask and the storage overpack may be performed at the unsheltered ISFSI concrete pad. It is therefore possible that the HI-STORM FW System (and/or the HI-TRAC VW transfer cask) may experience the extreme environmental conditions, resulting in the impact from a tornado-borne projectile.

#### 12.2.6.2 Tornado Analysis

A tornado event is characterized by high wind velocities and tornado-generated missiles. The reference missiles considered in this FSAR (see Section 2.2) are of three sizes: small, medium, and large. A small projectile, upon collision with a cask, would tend to penetrate it. A large projectile, such as an automobile, on the other hand, would tend to destabilize a free-standing cask. Accordingly, the tornado event has two distinct effects on the HI-STORM FW System. First, the tornado winds and/or tornado missile attempt to tip-over the loaded HI-STORM FW overpack or HI-TRAC VW transfer cask. The pressure loading of the high velocity winds and/or the impact of the large tornado missiles act to apply an overturning moment. The second effect is tornado missiles propelled by high velocity winds, which attempt to penetrate the HI-STORM FW overpack or HI-TRAC VW transfer cask.

During handling operations at the ISFSI pad, the loaded HI-TRAC VW transfer cask, while in the vertical orientation, shall be attached to a lifting device designed in accordance with the requirements specified in Subsection 2.3.3. Therefore, it is not credible that the tornado missile and/or wind could tip-over the loaded HI-TRAC VW transfer cask while it is being handled in the vertical orientation. Penetration by a small missile, however, is credible. The tornado wind and missile are assumed to act synergistically in the safety evaluation in Section 3.4 to determine the kinematic stability of the HI-STORM FW overpack.

##### i. Structural

Section 3.4 provides the analysis of the pressure loading which attempts to tip-over the HI-STORM FW 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 large tornado missiles.

Analyses provided in Section 3.4 also show that there is a potential for a tornado missile (8 inch steel cylinder) to penetrate the water jacket of the HI-TRAC VW transfer cask. The HI-STORM FW overpack will suffer minor local damages due to the missile impact with no significant damage in the shielding and there will be no damage to the MPC.

##### ii. Thermal

The thermal consequences of the complete loss of water due to rupture of the water jacket from a tornado missile has been analyzed in Section 4.6. It has been demonstrated that the consequences are within the short term fuel cladding and material temperature limits.

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### iii. Shielding

Since the structural evaluation shows that the tornado missiles may penetrate the HI-TRAC VW water jacket and cause loss of water, for a conservative estimate of the dose rates a complete loss of water in the water jacket is assumed and is bounded by the fire condition assumptions. This assumption results in an increase in the radiation dose rates however the shielding analysis results presented in Chapter 5 demonstrate that the requirements of 10CFR72.106 are not exceeded.

### iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

### v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

### vi. Radiation Protection

HI-STORM FW overpack: There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not impact the MPC. Since there is only a possibility of minimal reduction in localized shielding and there is no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

HI-TRAC VW transfer cask: There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not impact the MPC. Increases in the local dose rates as a result of the possible loss of water in the HI-TRAC VW transfer cask water jacket are evaluated in Chapter 5. Immediately after the tornado accident a radiological inspection of the HI-TRAC VW transfer cask shall be performed and temporary shielding shall be installed if necessary to limit the exposure to the site personnel.

#### 12.2.6.3 Tornado Dose Calculations

The tornado winds do not tip-over the loaded HI-STORM FW overpack; damage the shielding materials of the HI-STORM FW overpack or HI-TRAC VW transfer cask; or damage the MPC Confinement Boundary. There is no affect on the radiation dose as a result of the tornado winds.

A tornado missile may cause localized damage in the concrete radial shielding of the HI-STORM FW overpack. However, the damage will have a negligible effect on the site boundary dose. A tornado missile may penetrate the HI-TRAC VW transfer cask water jacket shell causing the loss of the neutron shielding (water). The effects of the tornado missile damage on the loaded HI-TRAC

VW transfer cask is bounded by the post-accident dose assessment performed in Chapter 5, which conservatively assumes complete loss of the water in the water jacket and the water jacket shell.

#### 12.2.6.4 Tornado Accident Corrective Action

Following exposure of the HI-STORM FW System to a tornado, the ISFSI owner shall perform a visual and radiological inspection of the overpack and/or HI-TRAC VW transfer cask.

Damage sustained by the overpack outer shell, concrete, or vent screens shall be inspected and may be repaired, if required, while in-service. The HI-STORM FW overpack may continue its service after appropriate restoration (reapplication of coatings etc.) if there is no significant increase in the measured dose rates (i.e., the overpack's shielding effectiveness is confirmed) and if the final visual inspection is satisfactory.

Damage sustained by the HI-TRAC VW transfer cask shall be inspected and repaired. As appropriate, temporary shielding around the HI-TRAC VW transfer cask shall be installed. If damage to the HI-TRAC VW transfer cask water jacket or HI-TRAC VW transfer cask body is extensive and/or radiological conditions require (based on dose rate measurements), the HI-TRAC VW transfer cask shall be unloaded in accordance with Chapter 9, prior to repair.

#### 12.2.7 Flood

##### 12.2.7.1 Cause of Flood

Many ISFSIs are located in flood plains susceptible to floods. Therefore, it is necessary for such ISFSIs to define a Design Basis Flood (DBF). 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.

A flood event is characterized by two parameters:

- a. flood water velocity
- b. flood height over the ISFSI pad as a function of time

The design basis flood (DBF) event for an ISFSI site should provide the maximum flood water velocity. The highest flood height, on the other hand, is not the governing condition for the flood event because of the vented construction of the overpack. The bottom vents in the HI-STORM FW overpack ensure that the flood level inside and outside the overpack will be equal. When the flood waters are high and the MPC is fully submerged then there is no short-term threat to the storage system because the MPC's heat rejection to water is far more efficient than the (normal condition) heat rejection to air. The most adverse flood condition, therefore, exists when the flood waters are high enough to block the inlet openings but no higher. In this scenario, the MPC surface has minimum submergence in water and the ventilation air is completely blocked. In fact, as the flood water begins to accumulate on the ISFSI pad, the air passage size in the inlet vents begins to get

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smaller. Therefore, the rate of floodwater rise with time is necessary to analyze the thermal-hydraulic problem. For the reference design basis flood (DBF) analysis in this FSAR, the flood waters are assumed to rise instantaneously to the height to block the inlet vents and stay at that precise elevation for 32 hours. For each ISFSI site subject to a DBF, the flood time-history will be incorporated in determining the acceptability of the flood event. The acceptance criteria are provided in Section 2.2.

The analysis results for the reference DBF are presented below.

#### 12.2.7.2 Flood Analysis

##### i. Structural

The flood accident affects the HI-STORM FW overpack structure in two ways. The flood water velocity acts to apply an overturning moment, which attempts to tip-over the loaded overpack. The flood accident affects the MPC by applying an external pressure.

Section 3.4 provides the analysis of the flowing floodwater 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 MPC for the accident condition external pressure (Table 2.2.1) is presented in Section 3.4 and the resulting stresses from this event are shown to be well within the allowable values.

##### ii. Thermal

As stated above, for a flood of sufficient height to allow the water to come into extensive contact with the MPC, there is no adverse effect on the thermal performance of the system. The thermal consequence of such a flood is an increase in the rejection of the decay heat. Because the storage overpack is ventilated, water from a submerged flood will enter the annulus between the MPC and the overpack. The water would provide cooling that would be an order of magnitude greater than that available in the air filled annulus (due to water's higher heat transfer coefficient).

The reference DBF that is most adverse to heat rejection is treated for 100% blockage of air inlets in Subsection 12.2.13. If the duration of the flood blockage exceeds the DBF blockage specified in Subsection 4.6.2 then a site specific evaluation shall be performed in accordance with the methodology presented in Chapter 4 and evaluated for compliance with Subsection 2.2.3 criteria.

##### iii. Shielding

There is no effect on the shielding performance of the system as a result of this event. The

flood water provides additional shielding that reduces radiation doses.

iv. 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.

v. 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.

vi. 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-STORM FW System.

#### 12.2.7.3 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.

#### 12.2.7.4 Flood Accident Corrective Action

As shown in the analysis of the flood accident, the HI-STORM FW System sustains no damage as a result of the flood, which is a short-duration event. At the completion of the flood, exposed surfaces may need debris and adherent foreign matter removal.

### 12.2.8 Earthquake

#### 12.2.8.1 Cause of Earthquake

The HI-STORM FW System may be employed at any reactor or ISFSI facility in the United States. It is possible that during the use of the HI-STORM FW System, the ISFSI may experience an earthquake. The earthquake event postulated for the ISFSI is referred to as the “Design Basis Earthquake” (DBE). The DBE is defined in Subsection 2.2.3.

### 12.2.8.2 Earthquake Analysis

The earthquake accident analysis evaluates the effects of a seismic event on the loaded HI-STORM FW overpack based on a static stability criteria discussed in Subsection 2.2.3.

Some ISFSI sites will have earthquakes that exceed the static stability limit specified in Subsection 2.2.3. For these high-seismic sites, a dynamic analysis shall be performed based on the methodology provided in Subsection 3.4.4.

#### i. Structural

The methodology for the evaluation of the earthquake consequences has been presented in Subsection 3.4.4. An earthquake is a vibratory event, which is fully described by an acceleration time-history. However, for “weak” earthquakes, a static equilibrium based calculation suffices. However, for stronger DBEs, a dynamic analysis is required. The results from the reference DBE analyzed in Subsection 3.4.4 are used to evaluate the safety case for the earthquake using the acceptance criteria in Section 2.2.

The following conclusions can be reached from the structural analysis of the HI-STORM FW System under the reference DBE event:

- a. The MPC Confinement Boundary remains unbreached.
- b. The HI-STORM FW overpack structure remains intact; i.e., the lid is not displaced.
- c. There is no physical damage to the HI-STORM FW overpack shielding concrete.
- d. The HI-STORM FW overpack does not tip over.

#### ii. Thermal

There is no effect on the thermal performance of the system as a result of this event.

#### iii. Shielding

There is no effect on the shielding performance of the system as a result of this event.

#### iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

#### v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

#### vi. 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 reference DBE does not affect the continued safe operation of the HI-STORM FW System.

#### 12.2.8.3 Earthquake Dose Calculations

Structural analysis of the earthquake accident shows that the loaded overpack will not tip over as a result of the specified seismic activity. Since the loaded overpack does not tip-over, there is no increase in radiation dose rates at the site boundary.

#### 12.2.8.4 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 displaced from their installed position or tipped over. In the unlikely event of a tip-over, the corrective actions shall be in accordance with Subsection 12.2.3.

### 12.2.9 100% Fuel Rod Rupture

This accident event postulates the non-mechanistic condition that all the fuel rods rupture and that the quantities of fission product gases and fill gas are released from the fuel rods into the MPC cavity consistent with ISG-5, Revision 1.

#### 12.2.9.1 Cause of 100% Fuel Rod Rupture

Through all credible accident conditions, the HI-STORM FW System maintains the spent nuclear fuel in an inert environment while maintaining the peak fuel cladding temperature below the required short-term temperature limits, thereby providing assurance of fuel cladding integrity. Therefore, there is no credible cause for 100% fuel rod rupture. This accident is presumably postulated in NUREG-1536 to evaluate the MPC confinement barrier for the maximum possible internal pressure based on the non-mechanistic failure of 100% of the fuel rods.

#### 12.2.9.2 100% Fuel Rod Rupture Analysis

The 100% fuel rod rupture accident has no containment consequences. The event does not change the reactivity of the stored fuel, the magnitude of the radiation source, the shielding capability of the system, or the criticality control features of the fuel basket; and does not challenge the structural integrity of the MPC.

i. Structural

The structural analysis provided in Chapter 3 evaluates the MPC Confinement Boundary under the accident condition design internal pressure limit set in Table 2.2.1. Calculations in Chapter 4 show that the accident internal pressure limit bounds the pressure from 100% fuel rod rupture.

ii. Thermal

The determination of the maximum accident pressure is provided in Chapter 4. The MPC internal pressure for the 100% fuel rod rupture condition is presented in Table 4.4.5, which is bounded by the design basis accident condition MPC internal pressure limit set in Table 2.2.1.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

vi. 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 event does not affect the safe operation of the HI-STORM FW System.

### 12.2.9.3 100% Fuel Rod Rupture Dose Calculations

The breach of fuel cladding postulated in this accident event does not result in any physical change to the storage system other than some release of gases and a limited quantity of solids (particulates) into the gaseous helium space. The amount of the radiation source remains unaffected. Hence, the radiation dose at the site boundary will not change perceptibly, i.e., there are no consequences to the site boundary dose.



#### 12.2.9.4 100% Fuel Rod Rupture Accident Corrective Action

As shown in the analysis of the 100% fuel rod rupture accident, the MPC Confinement Boundary is not damaged. The HI-STORM FW System is designed to withstand this accident and continue performing the safe storage of spent nuclear fuel under normal storage conditions. No corrective actions are required.

#### 12.2.10 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 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 non-credible.

#### 12.2.11 Explosion

##### 12.2.11.1 Cause of Explosion

An explosion within the protected area of an ISFSI is improbable since there are no explosive materials permitted within the site boundary. However, an explosion as a result of combustion of the fuel contained in cask transport vehicle is possible. As the fuel available for the explosion is limited in quantity the effects of an explosion on a reinforced structure are minimal. 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 3.1.

##### 12.2.11.2 Explosion Analysis

Any credible explosion accident is bounded by the accident external design pressure (Table 2.2.1) analyzed as a result of the flood accident water depth in Subsection 12.2.7 and the tornado missile accident of Subsection 12.2.6, because explosive materials are not stored within close proximity to the casks. The bounding analysis shows that the MPC and the overpack can withstand the effects of substantial accident external pressures without collapse or rupture.

An ISFSI where a credible explosion event produces a pressure wave greater than analyzed in Subsection 12.2.7 or an impactive load greater than considered in Subsection 12.2.6, shall be evaluated within the purview of §72.212. The results of the safety evaluation of the postulated limiting explosion in this subsection are as follows:

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i. Structural

The structural evaluations for the MPC accident condition external pressure and overpack pressure differential are presented in Section 3.4 and demonstrate that all stresses are within allowable values.

ii. Thermal

There is no effect on the thermal performance of the system as a result of this event.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. 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.

vi. 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 reference explosion accident does not affect the safe operation of the HI-STORM FW System.

#### 12.2.11.3 Explosion Dose Calculations

The reference bounding external pressure load has no effect on the HI-STORM FW Overpack and MPC. Therefore, no effect on the shielding, criticality, thermal or confinement capabilities of the HI-STORM FW System is experienced as a result of the explosion pressure load. The effects of explosion generated (reference) missiles on the HI-STORM FW System structure is bounded by the analysis of tornado generated missiles.

#### 12.2.11.4 Explosion Accident Corrective Action

The explosive overpressure caused by an explosion is bounded by the external pressure exerted by the flood accident. The external pressure from the flood is shown not to damage the HI-STORM FW

System. Following an explosion, the ISFSI owner shall perform a visual and radiological inspection of the overpack. If the outer shell or concrete is damaged as a result of explosion generated missiles, the overpack will be repaired as necessary.

## 12.2.12 Lightning

### 12.2.12.1 Cause of Lightning

As the HI-STORM FW 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.

### 12.2.12.2 Lightning Analysis

The HI-STORM FW System is a large metal/concrete cask stored in an unsheltered ISFSI. As such, it may be subject to lightning strikes. When the HI-STORM FW overpack is struck with lightning, the lightning will discharge through the steel shell of the overpack to the ground. Lightning strikes have high currents, but their duration is short (i.e., less than a second). The overpack outer shell is composed of conductive carbon steel and, as such, provides a direct path to the ground through the optional grounding cable.

The MPC provides the Confinement Boundary for the spent nuclear fuel. The effects of a lightning strike will be limited to the overpack. The lightning current will discharge into the overpack and directly into the ground. Therefore, the MPC will be unaffected.

#### i. Structural

There is no structural consequence as a result of this event.

#### ii. Thermal

There is no effect on the thermal performance of the system as a result of this event.

#### iii. Shielding

There is no effect on the shielding performance of the system as a result of this event.

#### iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

#### v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

## vi. 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-STORM FW System.

### 12.2.12.3 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.

### 12.2.12.4 Lightning Accident Corrective Action

The HI-STORM FW System will not sustain any damage from the lightning accident. There is no surveillance or corrective action required.

## 12.2.13 100% Blockage of Air Inlets

### 12.2.13.1 Cause of 100% Blockage of Air Inlets

This event is defined as a complete blockage of all bottom inlets. A blockage of all of the circumferentially arrayed inlets cannot be realistically postulated to occur at most sites. However, a flood, blizzard snow accumulation, tornado debris, or volcanic activity, where applicable, can cause a significant blockage.

### 12.2.13.2 100% Blockage of Air Inlets Analysis

The immediate consequence of a complete blockage of the air inlet openings is that the normal circulation of air for cooling the MPC is stopped. An amount of heat will continue to be removed by localized air circulation patterns in the overpack annulus and outlet opening, and the MPC will continue to radiate heat to the relatively cooler storage overpack. As the temperatures of the MPC and its contents rise, the rate of heat rejection will increase correspondingly. Under this condition, the temperatures of the HI-STORM FW overpack, the MPC and the stored fuel assemblies will rise as a function of time.

As a result of the large mass and correspondingly large thermal capacity of the HI-STORM FW overpack, it is expected that a significant temperature rise is only possible if the blocked condition is allowed to persist for an extended duration. This accident condition is, however, a short duration event that will be identified by the ISFSI staff, at worst, during scheduled periodic surveillance at the ISFSI site and corrected using the site's emergency response process.

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i. Structural

There are no structural consequences as a result of this event.

ii. Thermal

A thermal analysis is performed in Subsection 4.6.2 to determine the effect of a complete blockage of all inlets for an extended duration. For this event, both the fuel cladding and component temperatures remain below their short-term temperature limits. The MPC internal pressure for this event is evaluated in Subsection 4.6.2 and is bounded by the design basis internal pressure for accident conditions (Table 2.2.1).

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event, since the concrete temperatures do not exceed the short-term condition design temperature provided in Table 2.2.3.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. Confinement

There is no effect on the confinement function of the MPC as a result of this event.

vi. 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 the above evaluation, it is concluded that the 100% blockage of air inlets accident does not affect the safe operation of the HI-STORM FW System, as the ISFSI's emergency response process required to act to remove the blockage is the first priority activity.

#### 12.2.13.3 100% Blockage of Air Inlets Dose Calculations

As shown in the analysis of the 100% blockage of air inlets accident, the shielding capabilities of the HI-STORM FW overpack are unchanged because the peak concrete temperature does not exceed its short-term condition design temperature. The elevated temperatures will not cause the breach of the confinement system and the short term fuel cladding temperature limit is not exceeded. Therefore, there is no radiological impact.

#### 12.2.13.4 100% Blockage of Air Inlets Accident Corrective Action

Analysis of the 100% blockage of air inlet accident shows that the temperatures for cask system components and fuel cladding are within the accident temperature limits if the blockage is cleared within the maximum elapsed period between scheduled surveillance inspections. Upon detection of the complete blockage of the air inlet openings, the ISFSI owner shall activate its emergency response procedure to remove the blockage with mechanical and manual means as necessary. After clearing the overpack openings, the overpack shall be visually and radiologically inspected for any damage. If exit air temperature monitoring is performed in lieu of direct visual inspections, the difference between the ambient air temperature and the exit air temperature will be the basis for the assurance that the temperature limits are not exceeded.

For an accident event that completely blocks the inlet or outlet air openings for greater than the analyzed duration, a site-specific evaluation or analysis may be performed to whether adequate heat removal for the duration of the event would occur. Adequate heat removal is defined as the minimum rate of heat dissipation that ensures cladding temperatures limits are met and structural integrity of the MPC and overpack is not compromised. For those events where an evaluation or analysis is not performed or is not successful in showing that cladding temperatures remain below their short term temperature limits, the site's emergency plan shall include provisions to address removal of the material blocking the air inlet openings and to provide alternate means of cooling prior to exceeding the time when the fuel cladding temperature reaches its short-term temperature limit. Alternate means of cooling could include, for example, spraying water into the air outlet opening using pumps or fire-hoses or blowing air into the air outlet opening, to directly cool the MPC.

#### 12.2.14 Burial Under Debris

##### 12.2.14.1 Cause of Burial Under Debris

Complete burial of the HI-STORM FW System under debris is not a credible accident. During storage at the ISFSI, there are no structures above the casks that may collapse and surround them. The minimum regulatory distance(s) 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-STORM FW System to become completely buried under debris. However, for conservatism, complete burial under debris is considered. Blockage of the HI-STORM FW overpack air inlet openings has already been considered in Subsection 12.2.12.

##### 12.2.14.2 Burial Under Debris Analysis

Burial of the HI-STORM FW 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. A perverse effect of the overlaid debris would be to provide additional

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shielding to reduce radiation doses. The accident external pressure considered in this FSAR during the flood 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-STORM FW 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 limit during a burial under debris accident, assuming that the debris has the consistency of a typical pile of rocks (pebbles).

i. Structural

The structural evaluation of the MPC enclosure vessel for accident internal pressure conditions set in Table 2.1.1 bounds the pressure calculated for this event. Therefore, the resulting stresses from this event are well within the allowable values, as demonstrated in Section 3.4.

ii. Thermal

The fuel cladding and MPC integrity is evaluated in Subsection 4.6.2. The evaluation demonstrates that the fuel cladding and confinement function of the MPC are not compromised even if the burial event lasts for a substantial duration.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. 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.

vi. Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no deleterious effect on the site boundary dose a result of this event.

Based on the above evaluation, it is concluded that the burial under debris accident does not affect the safe operation of the HI-STORM FW System, as the ISFSI's emergency response process required to act to remove the debris is the first priority activity.

### 12.2.14.3 Burial Under Debris Dose Calculations

As discussed in burial under debris analysis, the shielding is enhanced while the HI-STORM FW System is covered.

The elevated temperatures will not cause the breach of the confinement system and the short term fuel cladding temperature limit is not exceeded. Therefore, there is no adverse radiological impact.

### 12.2.14.4 Burial Under Debris Accident Corrective Action

Analysis of the burial under debris accident shows that the fuel cladding peak temperatures are not exceeded even for an extended duration of burial. 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 storage overpack, the storage overpack shall be visually and radiologically inspected for any damage. The loaded MPC shall be removed from the storage overpack with the HI-TRAC VW transfer cask to allow complete inspection of the overpack air inlets and outlets, and annulus. Removal of obstructions to the air flow path shall be performed prior to the re-insertion of the MPC. The site's emergency action plan shall include provisions for the implementation of this corrective action.

## 12.2.15 Extreme Environmental Temperature

### 12.2.15.1 Cause of Extreme Environmental Temperature

The extreme environmental temperature is postulated (see Table 2.2.2) as a 3-day average temperature caused by extreme weather conditions.

### 12.2.15.2 Extreme Environmental Temperature Analysis

To determine the effects of the extreme temperature, it is conservatively assumed that the temperature persists for a sufficient duration (3 days) to allow the HI-STORM FW overpack to achieve thermal equilibrium.

The accident condition considering an extreme environmental temperature (Table 2.2.2) for a duration sufficient to reach thermal equilibrium is evaluated with respect to accident condition design temperatures listed in Table 2.2.3.

#### i. Structural

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.



ii. Thermal

The resulting temperatures for the system and fuel assembly cladding are provided in evaluation performed in Subsection 4.6.2. As concluded from this evaluation, all temperatures are within the short-term accident condition allowable values specified in Table 2.2.3.

iii. Shielding

There is no effect on the shielding performance of the system as a result of this event, since the concrete temperature does not exceed the short-term temperature limit specified in Table 2.2.3.

iv. Criticality

There is no effect on the criticality control features of the system as a result of this event.

v. 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.

vi. 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 extreme environment temperature accident does not affect the safe operation of the HI-STORM FW System.

#### 12.2.15.3 Extreme Environmental Temperature Dose Calculations

The extreme environmental temperature will not cause the concrete to exceed its normal design temperature. Therefore, there will be no degradation of the concrete's shielding effectiveness. The elevated temperatures will not cause a breach of the confinement system and the short-term fuel cladding temperature is not exceeded. Therefore, there is no radiological impact on the HI-STORM FW System for the extreme environmental temperature and the dose calculations are to the same as those for normal condition dose rates.

#### 12.2.15.4 Extreme Environmental Temperature Corrective Action

There are no consequences of this accident that require corrective action.

Table 12.2.1

## ACCIDENT EVENTS AND THEIR PROBABILITY OF OCCURRENCE

	Event	Probability of Occurrence	Subsection Where Addressed	Comments
1.	HI-TRAC VW Transfer Cask Handling Accident	Non-Credible	12.2.1	This FSAR mandates the use of high integrity handling equipment and single-failure-proof lifting devices for handling loaded HI-TRAC VWs within the Part 72 jurisdictional boundary.
2.	HI-STORM FW Overpack Handling Accident	Non-Credible	12.2.2	This FSAR mandates the use of high integrity handling equipment and single-failure-proof lifting devices for handling loaded HI-STORM FWs within the Part 72 jurisdictional boundary.
3.	HI-STORM FW Non-Mechanistic Tip-Over	Non-Credible	12.2.3	The HI-STORM FW tip-over event is more properly referred to as a “non-mechanistic” tip-over, meaning that no physical loading considered in this FSAR leads to a tip-over event.
4.	Fire	Very small probability but credible	12.2.4	Although there are no ignition sources in the ISFSI area, combustible material (motive fuel) is present. Therefore, the potential of a fire event cannot be ruled out categorically.
5.	Partial Blockage of MPC Basket Vent Holes	Non-Credible	12.2.5	An impactive event may jolt the stored fuel and cause its crud to fall off. However, as explained in Subsection 12.2.5, there is no realistic mechanism for the blockage of the flow holes.
6.	Tornado	Credible	12.2.6	Because a HI-STORM FW ISFSI can be deployed in any state within the U.S., the potential of a tornado event at a generic ISFSI must be considered.
7.	Flood	Credible	12.2.7	Flood, like tornado, must be categorized as a credible event at a generic ISFSI site.

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Table 12.2.1

## ACCIDENT EVENTS AND THEIR PROBABILITY OF OCCURRENCE

	<b>Event</b>	<b>Probability of Occurrence</b>	<b>Subsection Where Addressed</b>	<b>Comments</b>
8.	Earthquake	Credible	12.2.8	The Design Basis Earthquake for an ISFSI is a specified event for a nuclear facility.
9.	100% Fuel Rod Rupture	Non-Credible	12.2.9	“100% rod rupture” is a non-mechanistic event; no specific loading event has been identified to cause 100% rod rupture.
10.	Confinement Boundary Leakage	Non-Credible	12.2.10	The Confinement Boundary has been determined to be invulnerable to leakage in Chapter 7.
11.	Explosion	Credible	12.2.11	Explosion of gasoline is a credible event at an ISFSI.
12.	Lightning	Credible	12.2.12	Lightning is a small probability event at any ISFSI, hence, it cannot be deemed non-credible.
13.	100% Blockage of Air Inlets	Non-Credible	12.2.13	Because the air openings are along the circumference of the cask, and surveillance is at very short intervals (see Technical Specification), the assumption of blockage of all openings has no mechanistic basis.
14.	Burial Under Debris	Credible	12.2.14	Burial of a loaded system under a debris cannot be generically ruled out because a nuclear plant site may (ever so minimally) susceptible to a large adverse environment event such as a tsunami or an avalanche.
15.	Extreme Environmental Temperature	Credible	12.2.15	In certain desert areas in the country a temperature spike that reaches the accident temperature limit (Table 2.2.2) cannot be ruled out. Such areas have not been declared unfit at a nuclear plant site by the USNRC and therefore, must be factored in defining generic accident events.

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## 12.3 OTHER EVENTS

This section addresses the MPC reflood event which is placed in the category of “other events” since it cannot be categorized as an off-normal or accident event. The MPC reflood event occurs if an ISFSI owner needs to return the fuel in a loaded canister to wet storage in the plant’s fuel pool. The MPC reflood event is presented in Subsection 9.4.3 in connection with the preparation for the MPC unloading operation. The reflooding of a loaded MPC with water results in a change to the environment around the fuel from a gaseous (low heat transfer medium) to aqueous (high heat transfer medium). This implies the generation of thermal stresses in the fuel cladding and potential for loss of cladding integrity.

The safety analysis of the reflood event in Subsection 3.4.4 focuses on the effect of strains (due to reflooding) on the integrity of the cladding. This safety analysis, which uses an appropriate thermal/structural model as well as evaluations in Subsection 4.5.5, forms the basis for the instructions in the MPC reflood procedures provided in Subsection 9.4.3. For a complete evaluation of the effects of the MPC reflood event on the MPC and spent nuclear fuel, the postulated cause of the event, monitoring of the event, analysis of the event effects and consequences, corrective actions, and radiological impact from the event are presented in this section similar to the systematic evaluations presented for design basis off-normal and postulated accident condition events in the preceding sections.

The results of the evaluations performed herein demonstrate that the HI-STORM FW System can withstand the effects of MPC reflood and remain in compliance with the applicable acceptance criteria. In particular, the integrity of the fuel cladding shall be preserved. The following subsections contain the evaluation of the effects of the MPC reflood event on the MPC and spent nuclear fuel that demonstrates that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses continue to meet the requirements of 10CFR72.104(a) and 10CFR20.

### 12.3.1 MPC Reflood

MPC reflood is performed during the preparation for the unloading operations as described in Subsection 9.4.3. The MPC is flooded with water at a controlled rate as specified in Subsection 4.5.5 with the MPC vent port open such that the generation of steam from flashing of water is not excessive and the pressure within the MPC remains below its normal condition internal design pressure. Although past industry experience generally supports cooldown of cask internals and fuel from hot storage conditions by a direct introduction of water into the canister space (as specified in Subsection 9.4.3) a structural evaluation has been performed to ensure fuel cladding integrity and is provided in Paragraph 3.4.4.1.

### 12.3.1.1 Postulated Cause of MPC Reflood

Likely causes to perform MPC reflood include those associated with required actions for certain limiting conditions for operation (LCO) (as specified in Appendix A of the Technical Specifications) to unload fuel assemblies from the MPC. A reflood operation may also be carried out at the plant owner's volition to return the fuel to wet storage as a voluntary act.

### 12.3.1.2 Monitoring of MPC Reflood

MPC reflooding is monitored at frequent intervals by the surveillance of MPC pressure as required by SR 3.1.3.1 to Specification 3.1.3 (Appendix A of the Technical Specifications). An indication of MPC pressurization at or above normal condition MPC internal design pressure established in Section 2.2 requires the immediate action to stop reflooding operations until the MPC cavity pressure is below the required limit. See LCO 3.1.3 and the associated basis in Chapter 13 for more information.

### 12.3.1.3 Analysis of Effects and Consequences of MPC Reflood

#### i. Structural

MPC Enclosure Vessel Integrity: The MPC water reflood rate specified in Subsection 4.5.5, the essential reflooding control procedure steps established in Subsection 9.4.3, and the surveillance instructions in SR 3.1.3.1 Specification 3.1.3 (Appendix A of the Technical Specifications) ensure that the MPC is maintained below the normal condition pressure limit and well below MPC off-normal and accident condition pressure limits set down in Section 2.2, thus ensuring large margins of safety and no harmful effect on the MPC enclosure vessel integrity.

Fuel Cladding Integrity: The structural evaluation in Paragraph 3.4.4.1 ensures that the fuel cladding integrity is preserved during the reflood event.

Other Structural Related Considerations: MPC reflooding is performed under a specified maximum water injection rate and below normal condition MPC internal design pressure. The pressures and temperatures are therefore compatible with design limits of existing MPC ancillaries and standard connections such as RVOAs. Maintaining the pressure and temperature parameters well below the design basis values for the ancillaries ensures that failure of components and appurtenances during the reflooding operation is unlikely. Therefore, no credible mechanism for risk to the plant staff or general public from radiological release due to the reflood operation can be identified.

#### ii. Thermal

A thermal evaluation is provided in Subsection 4.5.5 to specify the maximum water reflood rate. The maximum calculated water reflood rate will prevent MPC over-pressurization and fuel cladding damage.

### iii. Shielding

There is no adverse effect on the shielding performance of the system as a result of the MPC reflood event. The shielding performance of the MPC is indeed enhanced by the flooding of its contents.

### iv. Criticality

There is no adverse effect on the criticality control of the system as a result of this planned plant event. The essential procedure steps in Chapter 9 and surveillance SR 3.3.1.1 to Specification 3.3.1 (Appendix A of the Technical Specifications) ensure that the water used to reflood the MPC will have the minimum required soluble boron concentration. The generation and escape of steam from the MPC will increase (not lower) the soluble boron concentration.

### v. Confinement

During the reflood operation, the MPC confinement function is inoperative (and supplanted by the part 50 facility) as the canister is connected to the plant's fluid accumulation system and the source of water (such as the fuel pool). The reflooding operation, however, does not degrade the confinement capability of the MPC because the internal pressures and temperatures are procedurally controlled to remain well within the design limits.

### vi. Radiation Protection

As there is no adverse effect on the shielding or confinement functions, there is no adverse effect on occupational or public exposures as a result of this MPC reflood event. The vent port steam is delivered to the radwaste gas facility of the plant in accordance with the specified procedure in Subsection 9.4.3.

Based on this evaluation, it is concluded that MPC reflood has no adverse effects or consequences on the safety or operability of the HI-STORM FW System.

#### 12.3.1.4 Corrective Action for MPC Reflood

See Specification 3.1.3 (Appendix A of the Technical Specifications) on MPC Cavity Reflooding and Specification 3.3.1 (Appendix A of the Technical Specifications) on Boron Concentration.

#### 12.3.1.5 Radiological Impact of MPC Reflood

The event has no radiological impact because the plant's confinement barrier and shielding infrastructure are unaffected and the operation relies on no new system for the control of effluents.

## 12.4 REFERENCES

Currently no references listed.

# CHAPTER 13<sup>†</sup>: OPERATING CONTROLS AND LIMITS

## 13.0 INTRODUCTION

The HI-STORM FW 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 single-purpose ventilated metal-concrete overpack. This chapter defines the operating controls and limits (i.e., Technical Specifications) including their supporting bases for deployment and storage of a HI-STORM FW system at an ISFSI. The information provided in this chapter is in full compliance with NUREG-1536 [13.1.1].

## 13.1 PROPOSED OPERATING CONTROLS AND LIMITS

### 13.1.1 NUREG-1536 (Standard Review Plan) Acceptance Criteria

This portion of the FSAR establishes the commitments regarding the HI-STORM FW system and its use. Other 10CFR72 [13.1.2] and 10CFR20 [13.1.3] requirements in addition to the Technical Specifications may apply. The conditions for a general license holder found in 10CFR72.212 [13.1.2] shall be met by the licensee prior to loading spent fuel into the HI-STORM FW system. The general license conditions governed by 10CFR72 [13.1.2] are not repeated within these Technical Specifications. Licensees are required to comply with all commitments and requirements.

The Technical Specifications provided in Appendix A to the CoC and the authorized contents and design features provided in Appendix B to the CoC are primarily established to maintain subcriticality, the confinement boundary, shielding and radiological protection, heat removal capability, and structural integrity under normal, off-normal and accident conditions. Table 13.1.1 addresses each of these conditions respectively and identifies the appropriate Technical Specification(s) designed to control the condition. Table 13.1.2 provides the list of Technical Specifications for the HI-STORM FW system.

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<sup>†</sup> 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. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the Glossary.



Table 13.1.1	
HI-STORM FW SYSTEM CONTROLS	
Condition to be Controlled	Applicable Technical Specifications <sup>†</sup>
Criticality Control	3.3.1 Boron Concentration
Confinement boundary integrity and integrity of cladding on undamaged fuel	3.1.1 Multi-Purpose Canister (MPC)
Shielding and radiological protection	3.1.1 Multi-Purpose Canister (MPC) 3.1.3 MPC Reflooding 3.2.1 TRANSFER CASK Surface Contamination 5.1 Radioactive Effluent Control Program 5.3 Radiation Protection Program
Heat removal capability	3.1.1 Multi-Purpose Canister (MPC) 3.1.2 SFSC Heat Removal System
Structural integrity	5.2 Transport Evaluation Program

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<sup>†</sup> Technical Specifications are located in Appendix A to the CoC. Authorized contents are specified in this FSAR in Subsection 2.1.8

Table 13.1.2	
HI-STORM FW SYSTEM 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	Not Used
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1	SFSC Integrity
3.1.1	Multi-Purpose Canister (MPC)
3.1.2	SFSC Heat Removal System
3.1.3	Fuel Cool-Down
3.2	SFSC Radiation Protection
3.2.1	TRANSFER CASK Surface Contamination
3.3	SFSC Criticality Control
3.3.1	Boron Concentration
Table 3-1	MPC Cavity Drying Limits
Table 3-2	MPC Helium Backfill Limits
4.0	Not Used
5.0	ADMINISTRATIVE CONTROLS
5.1	Radioactive Effluent Control Program
5.2	Transport Evaluation Program
5.3	Radiation Protection Program

## 13.2 DEVELOPMENT OF OPERATING CONTROLS AND LIMITS

This section provides a discussion of the operating controls and limits, and training requirements for the HI-STORM FW system to assure long-term performance consistent with the conditions analyzed in this FSAR.

### 13.2.1 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-STORM FW 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-STORM FW System Design (overview);
2. ISFSI Facility Design (overview);
3. Systems, Structures, and Components Important-to-Safety (overview);
4. HI-STORM FW System Safety Analysis Report (overview);
5. NRC Safety Evaluation Report (overview);
6. Certificate of Compliance conditions;
7. HI-STORM FW Technical Specifications, Approved Contents, Design Features and other Conditions for Use;
8. HI-STORM FW Regulatory Requirements (e.g., 10CFR72.48, 10CFR72, Subpart K, 10CFR20, 10CFR73);
9. Required instrumentation and use;
10. Operating Experience Reviews
11. HI-STORM FW System and ISFSI Procedures, including
  - Procedural overview
  - Fuel qualification and loading
  - MPC /HI-TRAC/overpack rigging and handling, including safe load pathways
  - MPC welding operations
  - HI-TRAC/overpack staging operation
  - Auxiliary equipment operation and maintenance (e.g., draining, moisture removal, helium backfilling, supplemental cooling (if used), and cooldown)

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- MPC/HI-TRAC/overpack pre-operational and in-service inspections and tests
- Transfer and securing of the loaded HI-TRAC/overpack onto the transport vehicle
- Transfer and offloading of the HI-TRAC/overpack
- Preparation of MPC/HI-TRAC/overpack for fuel unloading
- Unloading fuel from the MPC/HI-TRAC/overpack
- Surveillance
- Radiation protection
- Maintenance
- Security
- Off-normal and accident conditions, responses, and corrective actions

### 13.2.2 Dry Run Training

A dry run training exercise of the loading, closure, handling, and transfer of the HI-STORM FW system shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. Dry run training already performed successfully for the HI-STORM 100 System can be substituted for dry run steps applicable to HI-STORM FW. The dry run shall include, but is not limited to the following:

1. Receipt inspection of the HI-STORM FW System components.
2. Moving the MPC/HI-TRAC into the spent fuel pool.
3. Preparation of the MPC/HI-TRAC for fuel loading.
4. Selection and verification of specific fuel assemblies to ensure conformance.
5. Locating specific assemblies and placing assemblies into the MPC/HI-TRAC (using a dummy fuel assembly), including appropriate independent verification.
6. Remote installation of the MPC lid and removal of the MPC/HI-TRAC from the spent fuel pool.
7. MPC welding, NDE inspections, pressure testing, draining, moisture removal, and helium backfilling (for which a mockup MPC may be used).
8. Placement of the HI-STORM FW System at the ISFSI.

### 13.2.3 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-STORM FW system is completely passive during storage and requires no monitoring instruments. The user may choose to implement a temperature monitoring system to verify operability of the overpack heat removal system in accordance with Technical Specification Limiting Condition for Operation (LCO) 3.1.2.

### 13.2.4 Limiting Conditions for Operation (LCO)

Limiting Conditions for Operation (LCO) specify the minimum capability or level of performance that is required to assure that the HI-STORM FW system can fulfill its safety functions.

### 13.2.5 Equipment

The HI-STORM FW 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-STORM FW 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.

### 13.2.6 Surveillance Requirements

The analyses provided in this FSAR show that the HI-STORM FW system fulfills its safety functions, provided that the Technical Specifications and the Authorized Contents described in Subsection 2.1.8 are met. Surveillance requirements during loading, unloading, and storage operations are provided in the Technical Specifications.

### 13.2.7 Design Features

This subsection describes HI-STORM FW system design features that are Important to Safety. These features require design controls and fabrication controls. The design features, detailed in this FSAR and in Appendix B to the CoC, are established in specifications and drawings which are controlled through the quality assurance program. Fabrication controls and inspections to assure that the HI-STORM FW system is fabricated in accordance with the design drawings and the requirements of this FSAR are described in Chapter 10.

### 13.2.8 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.

### 13.2.9 HI-STORM FW Overpack

- a. HI-STORM overpack material mechanical properties and dimensions for structural integrity to provide protection of the MPC and shielding of the spent nuclear fuel assemblies during handling and storage operations.
- b. HI-STORM overpack material thermal properties and dimensions for heat transfer control.
- c. HI-STORM overpack material composition and dimensions for dose rate control.

### 13.2.10 HI-TRAC VW Transfer Cask

- a. HI-TRAC transfer cask 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-TRAC transfer cask material thermal properties and dimensions for heat transfer control.
- c. HI-TRAC transfer cask material composition and dimensions for dose rate control.

### 13.2.11 Verifying Compliance with Fuel Assembly Decay Heat, Burnup, and Cooling Time Limits

The examples below demonstrate how the user of the system can determine if fuel assemblies, including NFH, are acceptable for loading in either the MPC-37 or the MPC-89 in accordance with the allowable decay heat, burnup, and cooling time for the approved contents.

### Example 1

In this example, it will be assumed that the MPC-37 is being loaded with array/class 17x17A fuel in its regionalized loading pattern as shown in Figure 1.2.1 with heat loads from Table 1.2.3.

Table 13.2.1 provides four hypothetical fuel assemblies in the 17x17A array/class that will be evaluated for acceptability for loading in the MPC-37. The decay heat values and the fuel classification in Table 13.2.1 are determined by the user. The other information is taken from the fuel assembly and reactor operating records.

Fuel Assembly Number 1 is acceptable for storage in Pattern A, Region 2 of MPC-37. Fuel Assembly Number 1 is not acceptable for storage in Pattern A Region 1 or Region 3 because the total heat load of the fuel assembly and the non-fuel hardware exceeds the assembly decay heat limit for those regions. Fuel Assembly Number 1 cannot be stored in Pattern B since the total heat load of the assembly exceeds the assembly decay heat limit for all three regions.

Fuel Assembly Number 2 is not acceptable for loading. Fuel Assembly 2 is limited to the cell locations for DFCs in the MPC-37 (Figure 2.1.1). These cells, which are a subset of Region 3, have a decay heat limit lower than the decay heat of the assembly, for both Pattern A and B. This assembly will need additional cooling time (reduction in decay heat) to be acceptable for loading in the MPC-37.

Fuel Assembly Number 3 is acceptable for loading in Region 1 or Region 2, both Pattern A and B. The fuel assembly is limited to these locations due to the non-fuel hardware (Figure 2.1.5) and the total heat load of the fuel assembly and non-fuel hardware is less than the assembly decay heat limits for these regions.

Fuel Assembly Number 4 is not acceptable for loading in the MPC-37 because its cooling time is less than the minimum of 3 years. When the fuel assembly attains three years cooling time it can be reevaluated based on the decay heat.

### Example 2

In this example, it will be assumed that the MPC-89 is being loaded with array/class 10x10A fuel in its regionalized storage pattern as shown in Figure 1.2.2 with heat loads from Table 1.2.4.

Table 13.2.2 provides four hypothetical fuel assemblies in the 10x10A array/class that will be evaluated for acceptability for loading in the MPC-89. The decay heat values and the fuel classification in Table 13.2.2 are determined by the user. The other information is taken from the fuel assembly and reactor operating records.

Fuel Assembly Number 1 is acceptable for loading in the MPC-89. Fuel Assembly 1 is limited to the cell locations for DFCs in the MPC-89 (Figure 2.1.2). These cells, which are a subset of Region 3, have a decay heat limit higher than the decay heat of the assembly, therefore the

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assembly is acceptable for loading in the MPC-89, but it is limited to the cells depicted in Figure 2.1.2

Fuel Assembly Number 2 is not acceptable for loading in the MPC-89. Fuel Assembly 2 is limited to the cell locations for DFCs in the MPC-89 (Figure 2.1.2). These cells, which are a subset of Region 3, have a decay heat limit lower than the decay heat of the assembly. This assembly will need additional cooling time (reduction in decay heat) to be acceptable for loading in the MPC-89.

Fuel Assembly Number 3 is acceptable for loading in Regions 1, 2 or 3 of the MPC-89.

Fuel Assembly Number 4 is acceptable for loading in Region 2 of the MPC-89 only. The fuel assembly is limited to these locations due to the total heat load of the fuel assembly.

Table 13.2.1				
SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE (Array/Class 17x17A)				
FUEL ASSEMBLY NUMBER	1	2	3	4
INITIAL ENRICHMENT (WT. % <sup>235</sup> U)	3.0	3.2	4.3	4.5
FUEL ASSEMBLY BURNUP (MWD/MTU)	37100	35250	41276	55000
FUEL ASSEMBLY COOLING TIME (YEARS)	4.7	3.3	18.2	2.9
FUEL ASSEMBLY DECAY HEAT (KW)	1.01	1.45	0.4	2.08
NON-FUEL HARDWARE STORED WITH ASSEMBLY	BPRA	None	NSA	None
NFH DECAY HEAT (KW)	0.5	0	0.3	0
FUEL CLASSIFICATION	Undamaged	Damaged	Undamaged	Undamaged



Table 13.2.2				
SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE (Array/Class 10x10A)				
FUEL ASSEMBLY NUMBER	1	2	3	4
INITIAL ENRICHMENT (WT. % <sup>235</sup> U)	3.0	3.2	4.3	4.5
FUEL ASSEMBLY BURNUP (MWD/MTU)	37100	35250	41276	55000
FUEL ASSEMBLY COOLING TIME (YEARS)	4.7	3.3	18.2	7
FUEL ASSEMBLY DECAY HEAT (KW)	0.43	0.55	0.2	0.61
FUEL CLASSIFICATION	Damaged	Damaged	Undamaged	Undamaged

### 13.3 TECHNICAL SPECIFICATIONS

Technical Specifications for the HI-STORM FW system are provided in Appendix A to the Certificate of Compliance. Authorized Contents (i.e., fuel specifications) and Design Features are provided in Appendix B to the CoC. Bases applicable to the Technical Specifications are provided in the FSAR Appendix 13.A. The format and content of the HI-STORM FW 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" [13.3.1] was used as a guide in the development of the Technical Specifications and Bases.

### 13.4 REGULATORY EVALUATION

Table 13.1.2 lists the Technical Specifications for the HI-STORM FW system. The Technical Specifications are detailed in Appendix A to the Certificate of Compliance. Authorized Contents (i.e., fuel specifications) and Design Features are provided in Appendix B to the CoC.

The conditions for use of the HI-STORM FW system identify necessary Technical Specifications, limits on authorized contents (i.e., fuel), and design features to satisfy 10 CFR Part 72, and the applicable acceptance criteria have been satisfied. Compliance with these Technical Specifications and other conditions of the Certificate of Compliance provides reasonable assurance that the HI-STORM FW system will provide safe storage of spent fuel and is in compliance with 10 CFR Part 72, the regulatory guides, applicable codes and standards, and accepted practices.

### 13.5 REFERENCES

- [13.1.1] U.S. Nuclear Regulatory Commission, NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems*, Final Report, January 1997.
- [13.1.2] 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*."
- [13.1.3] U.S. Code of Federal Regulations, Title 10, *Energy*, Part 20, *Standards for Protection Against Radiation*."
- [13.3.1] Nuclear Management and Resources Council, Inc. – *Writer's Guide for the Restructured Technical Specifications*, NUMARC 93-03, February 1993.

**HI-STORM FW SYSTEM FSAR**

**APPENDIX 13.A**

**TECHNICAL SPECIFICATION BASES  
FOR THE HOLTEC HI-STORM FW MPC STORAGE SYSTEM**

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B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

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BASES

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LCOs	LCO 3.0.1, 3.0.2, 3.0.4, and 3.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 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 3.0.2	<p>LCO 3.0.2 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:</p> <ol style="list-style-type: none"> <li>Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and</li> <li>Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.</li> </ol>

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 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.

(continued)

BASES	
LCO 3.0.2 (continued)	<p>Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.</p> <p>The 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.</p>
LCO 3.0.3	<p>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.</p>
LCO 3.0.4	<p>LCO 3.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the HI-STORM FW System in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:</p> <ol style="list-style-type: none"> <li>Facility conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and</li> <li>Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in being required to exit the Applicability desired to be entered to comply with the Required Actions.</li> </ol> <p>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.</p>

(continued)

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BASES

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LCO 3.0.4 (continued) The provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the unloading of an SFSC.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.5 LCO 3.0.5 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 3.0.2 (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.

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.

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B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

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BASES

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SRs	SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
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SR 3.0.1	SR 3.0.1 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 3.0.2, constitutes a failure to meet an LCO.
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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 FW 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 3.0.2, prior to returning equipment to service. Upon completion of maintenance, appropriate post-maintenance testing is required. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. 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.

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BASES

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SR 3.0.2 SR 3.0.2 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 3.0.2 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 3.0.2 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 3.0.2 is not applicable."

As stated in SR 3.0.2, 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 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 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

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BASES

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SR 3.0.3 SR 3.0.3 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 3.0.2, 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 FW 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 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 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.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

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(continued)

BASES	
SR 3.0.3 (continued)	<p>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.</p> <p>Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.</p>
SR 3.0.4	<p>SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.</p> <p>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.</p> <p>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.</p> <p>However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a change in specified condition. When a system, subsystem, division, component, device, or variable is outside its specified limits, the associated SR(s) are not required to be performed per SR 3.0.1, 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 3.0.4 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 3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to specified condition changes.</p>

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BASES

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SR 3.0.4 (continued)      The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 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 3.0.4 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.

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Multi-Purpose Canister  
B 3.1.1

## B 3.1 SFSC Integrity

### B 3.1.1 Multi-Purpose Canister (MPC)

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#### BASES

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##### BACKGROUND

A TRANSFER CASK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the CoC. A lid is then placed on the MPC. The TRANSFER CASK and MPC are raised to the top of the spent fuel pool surface. The TRANSFER CASK and MPC are then moved into the 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 port cover plates and closure ring are installed and welded. Inspections are performed on the welds.

MPC cavity moisture removal using vacuum drying or forced helium dehydration is performed to remove residual moisture from the MPC cavity space after the MPC has been drained of water. If vacuum drying is used, 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 decay heat of the fuel.

If forced helium dehydration is used, the dry gas introduced to the MPC cavity through the vent or drain port absorbs the residual moisture in the MPC. This humidified gas exits the MPC via the other port and the absorbed water is removed through condensation and/or mechanical drying. The dried helium is then forced back to the MPC until the temperature acceptance limit is met.

After the completion of drying, the MPC cavity is backfilled with helium meeting the requirements of the CoC.

Backfilling of the MPC fuel cavity with helium promotes gaseous heat dissipation and the inert atmosphere protects the fuel cladding. Backfilling the MPC with helium in the required quantity 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.

(continued)

Multi-Purpose Canister  
B 3.1.1

<b>BASES</b>	
<b>APPLICABLE SAFETY ANALYSIS</b>	The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the confinement boundary of 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. The thermal analyses of the MPC assume that the MPC cavity is filled with dry helium of a minimum quantity to ensure the assumptions used for convection heat transfer are preserved. Keeping the backfill pressure below the maximum value preserves the initial condition assumptions made in the MPC over-pressurization evaluation.
<b>LCO</b>	A dry, helium filled, and sealed MPC establishes an inert heat removal environment necessary to ensure the integrity of the fuel cladding. 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.
<b>APPLICABILITY</b>	The dry, sealed, and inert atmosphere is required to be in place prior to TRANSPORT OPERATIONS to ensure both the confinement and heat removal mechanisms are 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 fuel.

(continued)

BASES	
ACTIONS	<p>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.</p> <p>A.1 If the cavity vacuum drying pressure or demister exit gas temperature limit has been determined not to be met prior to TRANSPORT 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 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.</p> <p>A.2 Once the quantity of moisture potentially left in the MPC cavity is determined, a corrective action plan shall be developed and actions initiated to the extent necessary to return the MPC to an analyzed condition. Since the quantity of moisture estimated 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 represent a long-term degradation concern, 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.</p>

(continued)

BASES	
ACTIONS (continued)	<p data-bbox="492 235 1430 596"> <b>B.1</b>            If the helium backfill quantity limit has been determined not to be met prior to TRANSPORT OPERATIONS, an engineering evaluation is necessary to determine the quantity of helium within the MPC cavity. Since too much or too little helium in the MPC during these modes represents a potential overpressure or heat removal degradation 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.         </p> <p data-bbox="492 638 1430 1142"> <b>B.2</b>            Once the quantity of helium 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 either by adding or removing helium or by demonstrating through analysis that all system limits will continue to be met. Since the quantity of helium estimated under Required Action B.1 can range over a broad scale, different recovery strategies may be necessary. Since elevated or reduced helium quantities existing in the MPC cavity represent a potential overpressure or heat removal degradation concern, corrective actions should be developed and implemented in a timely manner. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.         </p> <p data-bbox="492 1184 1430 1614"> <b>C.1</b>            If the helium leak rate limit has been determined not to be met prior to TRANSPORT OPERATIONS, an engineering evaluation is necessary to determine the impact of increased helium leak rate on heat removal and off-site dose. Since the HI-STORM FW OVERPACK is a ventilated system, any leakage from the MPC is transported directly to the environment. Since an increased helium leak rate represents a potential challenge to MPC heat removal and the off-site doses, reasonably rapid action is warranted. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.         </p>

(continued)



Multi-Purpose Canister  
B 3.1.1

BASES	
ACTIONS (continued)	<p>C.2 Once the consequences of the elevated leak rate from the MPC 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 the evaluation performed under Required Action C.1, different recovery strategies may be necessary. Since an elevated helium leak rate represents a challenge to heat removal rates and offsite doses, reasonably rapid action is required. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.</p> <p>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 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.</p>
SURVEILLANCE REQUIREMENTS	<p>SR 3.1.1.1 , SR 3.1.1.2, and SR 3.1.1.3</p> <p>The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment. Under certain conditions, cavity dryness may be demonstrated either by evacuating the cavity to a very low absolute pressure and verifying that the pressure is held over a specified period of time or by recirculating dry helium through the MPC cavity to absorb moisture until the gas temperature or dew point at the specified location reaches and remains below the acceptance limit for the specified time period. A low vacuum pressure or a demister exit temperature meeting the acceptance limit is an indication that the cavity is dry. Other conditions require the forced helium dehydration method of moisture removal to be used to provide necessary cooling of the fuel during drying operations.</p>

(continued)

Multi-Purpose Canister  
B 3.1.1

<b>BASES</b>	
<b>SURVEILLANCE REQUIREMENTS (continued)</b>	<p>Cooling provided by normal operation of the forced helium dehydration system ensures that the fuel cladding temperature remains below the applicable limits since forced recirculation of helium provides more effective heat transfer than that which occurs during normal storage operations.</p> <p>The conditions and requirements for drying the MPC cavity based on the burnup class of the fuel (moderate or high), heat load, and the applicable short-term temperature limit are given in the CoC/TS Appendix A, Table 3-1. The temperature limits and associated cladding hoop stress calculation requirements are consistent with the guidance in NRC Interim Staff Guidance (ISG) Document 11.</p> <p>Having the proper quantity of helium in the MPC ensures adequate heat transfer from the fuel to the fuel basket and surrounding structure of the MPC and precludes any overpressure event from challenging the normal, off-normal, or accident design pressure of the MPC.</p> <p>Meeting the helium leak rate limit prior to TRANSPORT OPERATIONS ensures there is adequate helium in the MPC for long term storage and that there is no credible effluent dose from the MPC.</p> <p>All of these surveillances must be successfully performed once, prior to TRANSPORT OPERATIONS to ensure that the conditions are established for SFSC storage which preserve the analysis basis supporting the MPC design.</p>
<b>REFERENCES</b>	<ol style="list-style-type: none"> <li>1. FSAR Chapters 1, 4, 7 and 9</li> <li>2. Interim Staff Guidance Document 11, Rev. 3</li> <li>3. Interim Staff Guidance Document 18, Rev. 1</li> </ol>

SFSC Heat Removal System  
B 3.1.2

B 3.1 SFSC Integrity

B 3.1.2 SFSC Heat Removal System

<b>BASES</b>	
<b>BACKGROUND</b>	<p>The SFSC Heat Removal System is a passive, air-cooled, convective heat transfer system that ensures heat from the MPC canister is transferred to the environs by the chimney effect. Relatively cool air is drawn into the annulus between the OVERPACK and the MPC through the inlet air ducts. The MPC transfers its heat from the canister surface to the air via natural convection. The buoyancy created by the heating of the air creates a chimney effect and the air is forced back into the environs through the outlet air ducts at the top of the OVERPACK.</p>
<b>APPLICABLE SAFETY ANALYSIS</b>	<p>The thermal analyses of the SFSC take credit for the decay heat from the spent fuel assemblies being ultimately transferred to the ambient environment surrounding the OVERPACK. Transfer of heat away from the fuel assemblies ensures that the fuel cladding and other SFSC component temperatures do not exceed applicable limits. Under normal storage conditions, the inlet and outlet air ducts are unobstructed and full air flow (i.e., maximum heat transfer for the given ambient temperature) occurs.</p> <p>Analyses have been performed for the complete obstruction of half, and all inlet air ducts. Blockage of half of the inlet air ducts reduces air flow through the OVERPACK annulus and decreases heat transfer from the MPC. Under this off-normal condition, no SFSC components exceed the short term temperature limits.</p> <p>The complete blockage of all inlet air ducts stops normal air cooling of the MPC. The MPC will continue to radiate heat to the relatively cooler OVERPACK. With the loss of normal air cooling, the SFSC component temperatures will increase toward their respective short-term temperature limits. None of the components reach their temperature limits over the duration of the analyzed event.</p>

(continued)

SFSC Heat Removal System  
B 3.1.2

<b>BASES</b>	
<b>LCO</b>	<p>The SFSC Heat Removal System must be verified to be operable to preserve the assumptions of the thermal analyses. Operability is defined as at least 50% of the inlet air ducts available for air flow (i.e., unblocked). Operability of the heat removal system ensures that the decay heat generated by the stored fuel assemblies is transferred to the environs at a sufficient rate to maintain fuel cladding and other SFSC component temperatures within design limits.</p> <p>The intent of this LCO is to address those occurrences of air duct blockage that can be reasonably anticipated to occur from time to time at the ISFSI (i.e., Design Event I and II class events per ANSI/ANS-57.9). These events are of the type where corrective actions can usually be accomplished within one 8-hour operating shift to restore the heat removal system to operable status (e.g., removal of loose debris).</p> <p>This LCO is not intended to address low frequency, unexpected Design Event III and IV class events (ANSI/ANS-57.9) such as design basis accidents and extreme environmental phenomena that could potentially block one or more of the air ducts for an extended period of time (i.e., longer than the total Completion Time of the LCO). This class of events is addressed site-specifically as required by Section 3.4.10 of Appendix B to the CoC.</p>
<b>APPLICABILITY</b>	<p>The LCO is applicable during STORAGE OPERATIONS. Once an OVERPACK containing an MPC loaded with spent fuel has been placed in storage, the heat removal system must be operable to ensure adequate dissipation of the decay heat from the fuel assemblies.</p>
<b>ACTIONS</b>	<p>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.</p>

(continued)

SFSC Heat Removal System  
B 3.1.2

BASES	
ACTIONS (continued)	A.1 Although the heat removal system remains operable, the blockage should be cleared expeditiously.
	B.1 If the heat removal system has been determined to be inoperable, it must be restored to operable status within eight hours. Eight hours is a reasonable period of time (typically, one operating shift) to take action to remove the obstructions in the air flow path.
	C.1 If the heat removal system cannot be restored to operable status within eight hours, the innermost portion of the OVERPACK concrete may experience elevated temperatures. Therefore, dose rates are required to be measured to verify the effectiveness of the radiation shielding provided by the concrete. This Action must be performed immediately and repeated every twelve hours thereafter to provide timely and continued evaluation of the effectiveness of the concrete shielding. As necessary, the system user shall provide additional radiation protection measures such as temporary shielding. The Completion Time is reasonable considering the expected slow rate of deterioration, if any, of the concrete under elevated temperatures.
	C.2.1 In addition to Required Action C.1, efforts must continue to restore cooling to the SFSC. Efforts must continue to restore the heat removal system to operable status by removing the air flow obstruction(s) unless optional Required Action C.2.2 is being implemented.
	This Required Action must be complete in 24 hours. The Completion Time is consistent with the thermal analyses of this event, which show that all component temperatures remain below their short-term temperature limits up to 32 hours after event initiation.
(continued)	

SFSC Heat Removal System  
B 3.1.2

BASES	
ACTIONS (continued)	<p>C.2.1 (continued) The Completion Time reflects the 8 hours to complete Required Action B.1 and the appropriate balance of time consistent with the applicable analysis results. The event is assumed to begin at the time the SFSC heat removal system is declared inoperable. This is reasonable considering the low probability of all inlet ducts becoming simultaneously blocked by trash or debris.</p> <p>C.2.2 In lieu of implementing Required Action C.2.1, transfer of the MPC into a TRANSFER CASK will place the MPC in an analyzed condition and ensure adequate fuel cooling until actions to correct the heat removal system inoperability can be completed. Transfer of the MPC into a TRANSFER CASK removes the SFSC from the LCO Applicability since STORAGE OPERATIONS does not include times when the MPC resides in the TRANSFER CASK.</p> <p>An engineering evaluation must be performed to determine if any concrete deterioration has occurred in the OVERPACK which prevents it from performing its design function. If the evaluation is successful and the air flow obstructions have been cleared, the OVERPACK heat removal system may be considered operable and the MPC transferred back into the OVERPACK. Compliance with LCO 3.1.2 is then restored. If the evaluation is unsuccessful, the user must transfer the MPC into a different, fully qualified OVERPACK to resume STORAGE OPERATIONS and restore compliance with LCO 3.1.2</p> <p>In lieu of performing the engineering evaluation, the user may opt to proceed directly to transferring the MPC into a different, fully qualified OVERPACK or place the TRANSFER CASK in the spent fuel pool and unload the MPC.</p> <p>The Completion Time of 24 hours reflects the Completion Time from Required Action C.2.1 to ensure component temperatures remain below their short-term temperature limits for the respective decay heat loads.</p>
	(continued)

SFSC Heat Removal System  
B 3.1.2

<b>BASES</b>	
<b>SURVEILLANCE REQUIREMENTS</b>	<p>SR 3.1.2</p> <p>The long-term integrity of the stored fuel is dependent on the ability of the SFSC to reject heat from the MPC to the environment. There are two options for implementing SR 3.1.2, either of which is acceptable for demonstrating that the heat removal system is OPERABLE.</p> <p>Visual observation that all inlet and outlet air ducts are unobstructed ensures that air flow past the MPC is occurring and heat transfer is taking place. Greater than 50% blockage of the total inlet or outlet air duct area renders the heat removal system inoperable and this LCO not met. 50% or less blockage of the total inlet or outlet air duct area does not constitute inoperability of the heat removal system. However, corrective actions should be taken promptly to remove the obstruction and restore full flow through the affected duct(s).</p> <p>As an alternative, for an OVERPACK with air temperature monitoring instrumentation installed in the outlet air ducts, the temperature rise between ambient and the OVERPACK air outlet may be monitored to verify operability of the heat removal system. Blocked inlet or outlet air ducts will reduce air flow and increase the temperature rise experienced by the air as it removes heat from the MPC. Based on the analyses, provided the air temperature rise is less than the limit stated in the SR, adequate air flow and, therefore, adequate heat transfer is occurring to provide assurance of long term fuel cladding integrity. The reference ambient temperature used to perform this Surveillance shall be measured at the ISFSI facility.</p> <p>The Frequency of 24 hours is reasonable based on the time necessary for SFSC components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of blockage of air ducts.</p>
<b>REFERENCES</b>	<ol style="list-style-type: none"> <li>1. FSAR Chapter 4</li> <li>2. ANSI/ANS 57.9-1992</li> </ol>

## B 3.1 SFSC INTEGRITY

## B 3.1.3 MPC Cavity Reflooding

BASES	
BACKGROUND	<p>In the event that an MPC must be unloaded, the TRANSFER CASK with its enclosed MPC is returned to the 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 100 psig design pressure. This is accomplished via direct measurement of the MPC gas pressure or via analysis.</p> <p>After ensuring the MPC cavity pressure meets the LCO limit, the MPC is then reflooded 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.</p> <p>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 large steam quantities during re-flooding.</p>
APPLICABLE SAFETY ANALYSIS	<p>The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the MPC in which the fuel assemblies are stored. Standard practice in the dry storage industry has historically been to directly reflood the storage canister with water. This standard practice is known not to induce fuel cladding failures.</p> <p>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 large quantities of steam during MPC reflooding. (Ref. 1).</p>

(continued)



MPC Cavity Reflooding  
B 3.1.3

<b>BASES</b>	
<b>LCO</b>	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.
<b>APPLICABILITY</b>	<p>The MPC cavity pressure is controlled during UNLOADING OPERATIONS after the TRANSFER CASK and integral MPC are back in the FUEL BUILDING and are no longer suspended from, or secured in, the transporter. Therefore, the MPC Reflood LCO does not apply during TRANSPORT OPERATIONS and STORAGE OPERATIONS.</p> <p>A note has been added to the APPLICABILITY for LCO 3.1.3 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.</p>
<b>ACTIONS</b>	<p>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.</p> <p>A</p> <p>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 re-flooding the MPC.</p> <p>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.</p>

(continued)

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MPC Cavity Reflooding  
B 3.1.3

<hr/> BASES <hr/>	
SURVEILLANCE REQUIREMENTS	<p><u>SR 3.1.3.1</u></p> <p>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, sufficient steam venting capacity exists during MPC re-flooding.</p> <p>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.</p>
REFERENCES	<hr/> 1. FSAR Chapters 3, 4, 9 and 12. <hr/>

TRANSFER CASK Surface Contamination  
B 3.2.1

B 3.2 SFSC Radiation Protection

B 3.2.1 TRANSFER CASK Surface Contamination

BASES	
BACKGROUND	A TRANSFER CASK is immersed in the spent fuel pool in order to load the spent fuel assemblies. As a result, the surface of the TRANSFER CASK may become contaminated with the radioactive material from the spent fuel pool water. This contamination is removed prior to moving the TRANSFER CASK to the ISFSI, or prior to transferring the MPC into the OVERPACK, whichever occurs first, 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 during MPC transfer and transportation using the TRANSFER CASK, are based on the assumption that the exterior surfaces of the TRANSFER CASK have been decontaminated. Failure to decontaminate the surfaces of the TRANSFER CASK could lead to higher-than-projected occupational doses.
LCO	Removable surface contamination on the TRANSFER CASK exterior surfaces and accessible surfaces of the MPC is limited to 1000 dpm/100 cm <sup>2</sup> from beta and gamma sources and 20 dpm/100 cm <sup>2</sup> 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 TRANSFER CASK loading process.

(continued)

TRANSFER CASK Surface Contamination  
B 3.2.1

BASES	
LCO (continued)	<p>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. LCO 3.2.1 requires removable contamination to be within the specified limits for the exterior surfaces of the TRANSFER CASK 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.</p> <p>A Note for this LCO indicates that the limits on surface contamination are not prescribed for the TRANSFER CASK if MPC TRANSFER is to occur inside the FUEL BUILDING, however plant radiation protection procedures should guide these limits for occupational dose considerations.</p>
APPLICABILITY	<p>The applicability is modified by a note that states that the LCO is not applicable to the TRANSFER CASK if MPC transfer operations occur inside the FUEL BUILDING. This is consistent with the intent of this LCO, which is to ensure loose contamination on the loaded TRANSFER CASK and MPC outside the FUEL BUILDING is within limits. If the MPC transfer is performed inside the FUEL BUILDING the empty TRANSFER CASK remains behind and is treated like any other contaminated hardware under the user's Part 50 contamination control program.</p> <p>Verification that the surface contamination is less than the limit in the LCO is performed during LOADING OPERATIONS. This occurs before TRANSPORT OPERATIONS, when the LCO is applicable. Measurement of surface contamination is unnecessary during UNLOADING OPERATIONS as surface contamination would have been measured prior to moving the subject TRANSFER CASK to the ISFSI.</p>

(continued)

TRANSFER CASK Surface Contamination  
B 3.2.1

<hr/> BASES <hr/>	
ACTIONS	<p>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. A subsequent use of the TRANSFER CASK that does not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.</p> <p><u>A.1</u></p> <p>If the removable surface contamination of a TRANSFER CASK or MPC, as applicable, which has been loaded with spent fuel is not within the LCO limits, action must be initiated to decontaminate the TRANSFER CASK or MPC and bring the removable surface contamination to within limits. The Completion Time of 7 days is appropriate given that sufficient time is needed to prepare for, and complete the decontamination once the LCO is determined not to be met.</p>
SURVEILLANCE REQUIREMENTS	<p>SR 3.2.1.1</p> <p>This SR verifies that the removable surface contamination on the TRANSFER CASK and/or 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 TRANSFER CASK or OVERPACK can be moved to the ISFSI without spreading loose contamination.</p>
REFERENCES	<ol style="list-style-type: none"> <li>1. FSAR Chapter 9</li> <li>2. NRC IE Circular 81-07.</li> </ol>

Boron Concentration  
B 3.3.1

### B 3.3 SFSC Criticality Control

#### B 3.3.1 Boron Concentration

BASES	
BACKGROUND	<p>A TRANSFER CASK 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 TRANSFER CASK and MPC are raised to the top of the spent fuel pool surface. The TRANSFER CASK and MPC are then moved into the 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 MPC closure ring are installed and welded. Inspections are performed on the welds.</p> <p>For those MPCs containing PWR fuel assemblies 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.</p>
APPLICABLE SAFETY ANALYSIS	<p>The spent nuclear fuel stored in the SFSC is required to remain subcritical (<math>k_{\text{eff}} \leq 0.95</math>) under all conditions of storage. The HI-STORM FW 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-37, 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.</p>

(continued)

Boron Concentration  
B 3.3.1

BASES	
LCO	<p>Compliance with this LCO ensures that the stored fuel will remain subcritical with a <math>k_{\text{eff}} \leq 0.95</math> while water is in the MPC. LCOs 3.3.1.a provides the minimum concentration of soluble boron required in the MPC water for the MPC-37. The amount of soluble boron is dependent on the initial enrichment of the fuel assemblies to be loaded in the MPC. Fuel assemblies with an initial enrichment less than or equal to 4.0 wt. % U-235 require less soluble boron than those with initial enrichments greater than 4.0 wt. % U-235. For initial enrichments greater than 4.0 wt. % U-235 and up to 5.0 wt. % U-225, interpolation is permitted to determine the required minimum amount of soluble boron.</p> <p>All fuel assemblies loaded into the MPC-37 are limited by analysis to maximum enrichments of 5.0 wt. % U-235.</p> <p>The LCO also requires that the minimum soluble boron concentration for the most limiting fuel assembly array/class and classification 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.</p>
APPLICABILITY	<p>The boron concentration LCO is applicable whenever an MPC-37 has at least one PWR fuel assembly in a storage location and water in the MPC.</p> <p>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.</p> <p>During UNLOADING OPERATIONS, the LCO is applicable when the MPC is reflooded with water. Note that compliance with SR 3.0.4 assures that the water to be used to flood the MPC is of the correct boron concentration to ensure the LCO is satisfied upon entering the Applicability.</p>

(continued)

BASES	
ACTIONS	<p>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.</p> <p>A.1 and A.2</p> <p>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. If the boron concentration of water in the MPC is less than its limit, all LOADING OPERATIONS, UNLOADING OPERATIONS or positive reactivity additions must be suspended immediately.</p> <p>A.3</p> <p>In addition to immediately suspending LOADING OPERATIONS, UNLOADING OPERATIONS and 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 is 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.</p> <p>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.</p>

(continued)



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BASES

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SURVEILLANCE  
REQUIREMENTS

## SR 3.3.1.1

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 MPC using two independent measurements to ensure the requirements of 10 CFR 72.124(a) are met. These two independent measurements will be repeated every 48 hours while the MPC is submerged in water or if water is to be added to or recirculated through the MPC.

For UNLOADING OPERATIONS, this means verifying the boron concentration in the source of borated water to be used to reflood the MPC within four hours of commencing reflooding operations and every 48 hours after until all the fuel is removed from the MPC. Two independent measurements will be taken to ensure the requirements of 10 CFR 72.124(a) are met. This ensures that when the LCO is applicable (upon introducing water into the MPC), the LCO will be met.

Surveillance Requirement 3.3.1.1 is modified by a note which states that SR 3.3.1.1 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 premise 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. After the completion of the surveillance methods, events which might change the soluble boron concentration will be administratively controlled per the LCO. If actions are taken that could result in a reduction in the boron concentration the surveillance will be performed again.

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 closure welding and other MPC storage preparation activities are taking place in an elevated radiation area atop the MPC. Plant procedures should ensure

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Boron Concentration  
B 3.3.1

BASES	
SURVEILLANCE REQUIREMENTS	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.

# CHAPTER 14<sup>†</sup>: QUALITY ASSURANCE PROGRAM

## 14.0 INTRODUCTION

### 14.0.1 Overview

This chapter provides a summary of the quality assurance program implemented by Holtec International 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-STORM FW system which includes the HI-TRAC VW transfer cask. 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-STORM FW system are controlled under the NRC-approved Holtec Quality Assurance Program. The Holtec QA program manual [14.0.1] is approved by the NRC [14.0.2] 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 the assurance that activities performed in support of the supply of safety-significant items and services are performed correctly and in compliance with the procurement documents.

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<sup>†</sup> This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61.

## 14.0.2 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 14.0.1.

NUREG/CR-6407 [14.0.3] 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-STORM FW system and HI-TRAC VW transfer cask. The ITS categories assigned to the HI-STORM FW cask components are identified in Tables 2.0.1 through 2.0.8. Quality categories for ancillary equipment are provided in Chapter 9 of this FSAR. Quality categories for other equipment needed to deploy the HI-STORM FW 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 [14.0.3].

Activities affecting quality are defined by the purchaser's procurement contract for use of the HI-STORM FW 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-STORM FW 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 [14.0.4]. 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 and has accordingly been added to the list of deviations in Table 1.0.3. 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.

## 14.1 REFERENCES

- [14.0.1] Holtec International Quality Assurance Program, Latest Approved Revision.
- [14.0.2] NRC QA Program Approval for Radioactive Material Packages No. 0784, Docket 71-0784.
- [14.0.3] NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," February 1996.
- [14.0.4] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997.