HOLTEC INTERNATIONAL

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October 13, 2011

Mr. John Goshen
U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: Holtec International HI-STORM Flood/Wind Multipurpose Canister Storage System, USNRC Docket No. 72-1032, License Amendment Request 1032-1

Reference: Holtec Project 5018
LAR 1032-1

Dear Mr. Goshen:

Holtec International herein submits a request to amend Certificate of Compliance (CoC) 72-1032 for the Company's HI-STORM FW MPC Storage System. This license amendment request (LAR) 1032-1 seeks to:

1) add a new heat load pattern for the MPC-37;
2) broaden the backfill pressure range for both MPC-37 and MPC-89; and
3) update certain definitions related to fuel classification.

A summary of the proposed changes (SOPC), with more detailed description of the change, reason for the change and justification for the changes, is provided in Attachment 1. Attachment 1 also contains some clarification and editorial suggestions for the CoC/TS for your consideration. A copy of the CoC/TS is provided in Attachment 2. Attachment 3 provides the proposed FSAR Sections (Revision 0.A) which contain the results of the detailed analysis and evaluations in support of the LAR. All proposed changes in Attachments 2 and 3 are provided in mark-up format to facilitate the Staff's review.

We respectfully request that the technical review and the approval process for this amendment be completed by mid-2012 to support an effective amendment date by the end of 2012. We believe that the above schedule is attainable given the limited scope of the LAR.

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If you have any questions please feel free to contact me at 856-797-0900 x687.

Sincerely,

Tammy S. Morin
Licensing Manager
Holtec International

cc: Ms. Vonna Ordaz, USNRC (letter only)
Mr. Doug Weaver, USNRC (letter only)
Mr. Michael Waters, USNRC (letter only)
Holtec Group 1 (letter only via email)
HUG distribution (letter only via email)

Attachment 1: Summary of Proposed Changes (4 pages)
Attachment 2: Proposed Revised CoC/TS Changes in mark-up format (74 pages)
Attachment 3: Proposed FSAR Changes in mark-up format (253 pages) – Includes Sections 1.2, 2.1, 2.3, Chapter 4, Sections 5.0, 5.1, 5.2, 5.4, 6.1, 6.4, and Chapter 11 only.

Attachment 1 to Holtec Letter 5018019

LAR 1032-1, REVISION 0

SUMMARY OF PROPOSED CHANGES

Proposed Change #1

A new decay heat load pattern, "Pattern B", is proposed for the MPC-37. Changes made to the HI-STORM FW MPC Storage System CoC/TS are summarized as follows:

> CoC; Change Amendment Number, Effective Date and Licensing Chief to "TBD".
>
> Appendix A; Change Amendment Number to "TBD".
>
> Appendix A, Table 3-2; updated to reflect the backfill requirements for MPC-37 Pattern B. See also Proposed Change 2.
>
> Appendix B; Change Amendment Number to "TBD".
>
> Appendix B, Section 2.3; MPC-37 "Pattern B" decay heat limits for the cells/regions/total MPC are added to Table 2.3-1.

Reason for Proposed Change #1

The current decay heat load pattern for the MPC-37 allows loading assemblies with decay heat ranging from 0.97 kW to 1.78 kW in discrete cells/regions. In some cases this may be limiting to the users of the system; therefore the new pattern is proposed to allow additional loading flexibility in the MPC-37.

Justification for Proposed Change #1

Analyses and evaluations support the inclusion of this decay heat load pattern for the MPC-37 model in the HI-STORM FW Dry Storage Cask System Certificate of Compliance as modified and provided in Attachment 2. Attachment 3 provides the supporting information that will be added to the FSAR upon approval of this amendment. The proposed FSAR changes include the description of the models, analysis, and results pertaining to the decay heat load limits for MPC-37 Pattern B, as applicable.

Thermal Justification

Thermal analyses have been performed for this proposed decay heat load pattern, MPC-37 Pattern B. The total heat load, resulting PCT, and basket temperatures for long term storage are bounded by the already approved MPC-37 decay heat load pattern, now referred to as "Pattern A". Proposed FSAR Chapter 4 has been updated to discuss Pattern B. No changes are made to the already approved MPC-89 heat load pattern and no changes are necessary to the patterns for MPC-37 and MPC-89 for vacuum drying of high burnup fuel.

Structural Justification

Since the temperatures of the fuel and the MPC-37 basket are bounded by Pattern A and all the HI-STORM FW System components remain below their temperature and pressure design limits, there is no effect on the currently approved structural analysis.

Attachment 1 to Holtec Letter 5018019

LAR 1032-1, REVISION 0

SUMMARY OF PROPOSED CHANGES

Shielding Justification

The basis for protection against direct radiation for normal operating conditions to a real individual beyond the controlled area boundary is the limit specified in 10 CFR 72.104 (a). This limit is dependent on plant operations as well as site specific conditions; therefore the determination and comparison of ISFSI dose to this limit is necessarily site-specific. 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 10 CFR 72.212.

The HI-STORM FW overpack is designed to limit the calculated surface dose rates on the cask for all MPC designs. These design objective dose rates are given in FSAR Table 2.3.2. TS Appendix A, Section 5.3 sets dose rate limits for the side surface, excluding the inlet and outlet vents, and top center of the HI-STORM at 300 mrem/hr and 30 mrem/hr, respectively. In Chapter 5 of the proposed FSAR, the calculated HI-TRAC VW and HI-STORM FW overpack dose rates for the MPC-37 are updated to reflect Pattern B since this is the bounding case. The dose rates at the inlet vents (dose location 1) slightly exceed the current design objective dose rates in FSAR Table 2.3.2; however they do not exceed the limits in the TS for the side dose rate. The limits in proposed FSAR Table 2.3.2 are updated to be consistent with the current limits in the TS.

The HI-TRAC VW dose rates for MPC-37 Pattern B, based on the minimum nominal steel and lead thickness, remain below the TS limit.

The radiation protection chapter (proposed FSAR Chapter 11) has been updated considering the slight increase in estimated dose rates for the MPC-37 Pattern B.

ALARA discussions in Section 5.1 have been updated to highlight to the users of the HI-STORM FW System that while the new MPC-37 Pattern B allows assemblies with higher heat loads and therefore higher source terms in the outer region of the MPC, the guiding principle in selecting fuel should still be to preferentially place assemblies with higher source terms in the inner regions of the basket as far as reasonably possible.

Confinement Justification

Since the temperatures of the fuel and the basket are bounded by Pattern A and all the HI-STORM FW System components remain below their temperature and pressure design limits, there is no effect on the currently approved confinement analysis.

Criticality Justification

Since the criticality analyses do not utilize or credit any of the heat load loading configurations, there is no effect on the currently approved criticality analysis.

SUMMARY OF PROPOSED CHANGES

Proposed Change #2

Revise TS Appendix A, Table 3-2 for helium backfill requirements for the MPC-37 and MPC-89 heat load patterns.

Reason for Proposed Change #2

The current helium backfill pressure bands required for the higher heat load canisters (100% of design basis heat load for MPC37 and MPC-89) are restrictive (3 psi) when considering instrument uncertainties. To address this, pressure bands are increased for MPC-89 and MPC-37 Pattern A at 100% design basis heat load. Also, to allow more flexibility to the user, back fill pressure bands are provided for MPC-37 Pattern A for 90% of the MPC heat load (per assembly basis) in addition to the 80% and 100% already in the TS. The helium backfill pressure range for MPC-37 Pattern B is larger than that for MPC-37 Pattern A and is therefore only provided at 100% design basis heat load.

Justification for Proposed Change #2

Thermal analyses have been performed for the increased backfill pressure ranges and shows that PCT is unaffected for normal long term storage, short-term operations and accident conditions. In addition MPC internal pressure for normal long term storage, short-term operations and accident conditions remains below the design basis limits; therefore there is no effect on the current structural or confinement analysis. This change has no effect on the current criticality analysis which is bounded by the flooded MPC condition inside the HI-TRAC in all cases. The specific thermal analysis and results are discussed in the proposed FSAR Chapter 4. This change has no effect on the already approved structural, confinement, or shielding analysis.

Proposed Change #3

Modify definition of UNDAMAGED FUEL ASSEMBLY. Add definition of GROSSLY BREACH SPENT FUEL ROD and REPAIRED/RECONSTITUTED FUEL ASSEMBLY. This is to extend undamaged fuel to include low enriched and channeled BWR with potential cladding defects larger than pinhole leaks or hairline cracks but without gross breaches, and to clarify that repaired or reconstituted assemblies are also covered by the undamaged fuel definition.

Appendix A, Section 1.1 Definitions; the definition of UNDAMAGED FUEL ASSEMBLY is modified in the table.

Appendix A, Section 1.1 Definitions; a definition of GROSSLY BREACHED SPENT FUEL ROD is added to the table.

Appendix A, Section 1.1 Definitions; a definition of REPAIRED/RECONSTITUTED FUEL ASSEMBLY is added to the table.

Appendix B, Table 2.1-3; Note 14 is added.

LAR 1032-1, REVISION 0

SUMMARY OF PROPOSED CHANGES

Reason for Proposed Change #3

Modifying the definition of undamaged fuel, and adding definitions for grossly breached spent fuel rod and repaired/reconstituted fuel assembly provides further clarity to the user and consistency with the guidance on classifying fuel given in ISG-1 Revision 2. In addition, these definitions will serve some BWR users who have older, low enriched, channeled BWR fuel. With this proposed change they can load the fuel without damaged fuel containers and without restriction on amount or location in the MPC-89.

Justification for Proposed Change #3

ISG-1 Revision 2 provides functional/performance based guidance for classifying fuel as either undamaged or damaged. In accordance with the ISG, undamaged fuel may contain some cladding defects, as long as the fuel is protected from high temperatures and oxidation, and does not contain gross cladding breaches. The HI-STORM FW System requires backfilling with helium and is shown to keep the peak cladding temperature of the fuel below the limits in ISG-11 Revision 3; therefore fuel is protected during storage against conditions that would lead to gross ruptures. As long as the fuel does not already contain a gross breach, there is no means to release fragments during storage. In addition, fuel that contains an assembly defect may be considered undamaged per ISG-1 Revision 2 if it can still meet fuel-specific and system related functions; therefore repaired and/or reconstituted assemblies, as proposed in the definition as part of this change, are considered undamaged.

For channeled BWR fuel, inspections to classify the fuel cladding as undamaged in accordance with the currently approved definition may be prohibitive from a cost, ALARA, or safety perspective. A particular subset of fuel, as described and analyzed in proposed FSAR Subsection 6.4.9, is shown to remain subcritical even if there was significant cladding damage and rearrangement of the fuel rods inside the channel. Therefore, if it can be determined that this fuel does not have gross cladding breaches, can be handled by normal means, and has enrichment less than or equal to 3.3 wt%, then it does not require a damaged fuel container nor is it limited to certain basket locations in the MPC.

Clarifications and Editorial Suggestions

The following are provided in the marked up documents for Staff's consideration.

Appendix A, TOC

Appendix A, Section 1.0, page numbering

Appendix A, Table 3-1, Note 4

Appendix B, TOC

Appendix B, Table 2.1-3, Notes 12 and 13

NRC FORM 651 (10-2004) 10 CFR 72

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS

The U.S. Nuclear Regulatory Commission is issuing this Certificate of Compliance pursuant to Title 10 of the Code of Federal Regulations, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR Part 72). This certificate is issued in accordance with 10 CFR 72.238, certifying that the storage design and contents described below meet the applicable safety standards set forth in 10 CFR Part 72, Subpart L, and on the basis of the Final Safety Analysis Report (FSAR) of the cask design. This certificate is conditional upon fulfilling the requirements of 10 CFR Part 72, as applicable, and the conditions specified below.

Certificate No.	Effective Date	Expiration Date	Docket No.	Amendment No.	Amendment Effective Date	Package Identification No.
1032	*TBD*~~June 13, 2011~~	June 12, 2051	72-1032	*TBD*~~0~~		USA/72-1032

Issued To: (Name/Address)

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

Safety Analysis Report Title

Holtec International
Final Safety Analysis Report for the
HI-STORM FW MPC Storage System

This certificate is conditioned upon fulfilling the requirements of 10 CFR Part 72, as applicable, the attached Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), and the conditions specified below:

APPROVED SPENT FUEL STORAGE CASK

Model No.: HI-STORM FW MPC Storage System

DESCRIPTION:

The HI-STORM FW MPC Storage System consists of the following components: (1) interchangeable multi-purpose canisters (MPCs), which contain the fuel; (2) a storage overpack (HI-STORM FW), which contains the MPC during storage; and (3) a transfer cask (HI-TRAC VW), which contains the MPC during loading, unloading and transfer operations. The MPC stores up to 37 pressurized water reactor fuel assemblies or up to 89 boiling water reactor fuel assemblies.

The HI-STORM FW MPC Storage System is certified as described in the Final Safety Analysis Report (FSAR) and in the U. S. Nuclear Regulatory Commission's (NRC) Safety Evaluation Report (SER) accompanying the Certificate of Compliance (CoC).

The MPC is the confinement system for the stored fuel. It is a welded, cylindrical canister with a honeycombed fuel basket, a baseplate, a lid, a closure ring, and the canister shell. All MPC components that may come into contact with spent fuel pool water or the ambient environment are made entirely of stainless steel or passivated aluminum/aluminum alloys. The canister shell, baseplate, lid, vent and drain port cover plates, and closure ring are the main confinement boundary components. All confinement boundary components are made entirely of stainless steel. The honeycombed basket provides criticality control.

DESCRIPTION (continued)

There are two types of MPCs: the MPC-37 and MPC-89. The number suffix indicates the maximum number of fuel assemblies permitted to be loaded in the MPC. Both MPC models have the same external diameter.

The HI-TRAC VW transfer cask provides shielding and structural protection of the MPC during loading, unloading, and movement of the MPC from the cask loading area to the storage overpack. The transfer cask is a multi-walled (carbon steel/lead/carbon steel) cylindrical vessel with a neutron shield jacket attached to the exterior and a retractable bottom lid used during transfer operations.

The HI-STORM FW storage overpack provides shielding and structural protection of the MPC during storage. The overpack is a heavy-walled steel and concrete, cylindrical vessel. Its side wall consists of plain (un-reinforced) concrete that is enclosed between inner and outer carbon steel shells. The overpack has air inlets at the bottom and air outlets at the top to allow air to circulate naturally through the cavity to cool the stored MPC. The inner shell has supports attached to its interior surface to guide the MPC during insertion and removal and provide a means to protect the MPC confinement boundary against impactive or impulsive loadings. A loaded MPC is stored within the HI-STORM FW storage overpack in a vertical orientation.

CONDITIONS

1. OPERATING PROCEDURES

Written operating procedures shall be prepared for handling, loading, movement, surveillance, and maintenance. The user's site-specific written operating procedures shall be consistent with the technical basis described in Chapter 9 of the FSAR.

2. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Written acceptance tests and a maintenance program shall be prepared consistent with the technical basis described in Chapter 10 of the FSAR. At completion of welding the MPC shell to baseplate, an MPC confinement weld helium leak test shall be performed using a helium mass spectrometer. The confinement boundary welds leakage rate test shall be performed in accordance with ANSI N14.5 to "leaktight" criterion. If a leakage rate exceeding the acceptance criteria 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.

3. QUALITY ASSURANCE

Activities in the areas of design, purchase, fabrication, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems and components, and decommissioning that are important-to-safety shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 72, Subpart G, and which is established, maintained, and executed with regard to the storage system

4. HEAVY LOADS REQUIREMENTS

Each lift of an MPC, a HI-TRAC VW transfer cask, or any HI-STORM FW overpack must be made in accordance to the existing heavy loads requirements and procedures of the licensed facility at which the lift is made. A plant-specific review of the heavy load handling procedures (under 10 CFR 50.59 or 10 CFR 72.48, as applicable) is required to show operational compliance with existing plant specific heavy loads requirements. Lifting operations outside of structures governed by 10 CFR Part 50 must be in accordance with Section 5.2 of Appendix A.

5. APPROVED CONTENTS

Contents of the HI-STORM FW MPC Storage System must meet the fuel specifications given in Appendix B to this certificate.

6. DESIGN FEATURES

Features or characteristics for the site or system must be in accordance with Appendix B to this certificate.

7. CHANGES TO THE CERTIFICATE OF COMPLIANCE

The holder of this certificate who desires to make changes to the certificate, which includes Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), shall submit an application for amendment of the certificate.

8. SPECIAL REQUIREMENTS FOR FIRST SYSTEMS IN PLACE

The air mass flow rate through the cask system will be determined by direct measurements of air velocity in the overpack cooling passages for the first HI-STORM FW MPC Cask System placed into service by any user with a heat load equal to or greater than 30 kW. The velocity will be measured in the annulus formed between the MPC shell and the overpack inner shell. An analysis shall be performed that demonstrates the measurements validate the analytic methods and thermal performance predicted by the licensing-basis thermal models in Chapter 4 of the FSAR.

A letter report summarizing the results of the thermal validation test and analysis shall be submitted to the NRC in accordance with 10 CFR 72.4. Cask users may satisfy this requirement by referencing a validation test report submitted to the NRC by another cask user.

9. PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

A dry run training exercise of the loading, closure, handling, unloading, and transfer of the HI-STORM FW MPC Storage System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The training exercise shall not be conducted with spent fuel in the MPC. The dry run may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The dry run shall include, but is not limited to the following:

a. Moving the MPC and the transfer cask into the spent fuel pool or cask loading pool.

b. Preparation of the HI-STORM FW MPC Storage System for fuel loading.

c. Selection and verification of specific fuel assemblies to ensure type conformance.

d. Loading specific assemblies and placing assemblies into the MPC (using a dummy fuel assembly), including appropriate independent verification.

e. Remote installation of the MPC lid and removal of the MPC and transfer cask from the spent fuel pool or cask loading pool.

f. MPC welding, NDE inspections, pressure testing, draining, moisture removal (by vacuum drying or forced helium dehydration, as applicable), and helium backfilling. (A mockup may be used for this dry-run exercise.)

g. Transfer of the MPC from the transfer cask to the overpack.

h. Placement of the HI-STORM FW MPC Storage System at the ISFSI.

i. HI-STORM FW MPC Storage System unloading, including flooding MPC cavity and removing MPC lid welds. (A mockup may be used for this dry-run exercise.)

Any of the above steps can be omitted if they have already been successfully carried out at a site to load a HI-STORM 100 System (USNRC Docket 72-1014).

10. AUTHORIZATION

The HI-STORM FW MPC Storage System, which is authorized by this certificate, is hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, this certificate, and the attached Appendices A and B. The HI-STORM FW MPC Storage System may be fabricated and used in accordance with any approved amendment to CoC No. 1032 listed in 10 CFR 72.214. Each of the licensed HI-STORM FW MPC Storage System components (i.e., the MPC, overpack, and transfer cask), if fabricated in accordance with any of the approved CoC Amendments, may be used with one another provided an assessment is performed by the CoC holder that demonstrates design compatibility. The HI-STORM FW MPC Storage System may be installed on an ISFSI pad with the HI-STORM 100 Cask System (USNRC Docket 72-1014) provided an assessment is performed by the CoC holder that demonstrates design compatibility.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

/RA/

~~Kimberly J. Hardin, Acting Chief~~*TBD*
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards
Washington, DC 20555

Dated ~~July 14, 2011~~*TBD*

Attachments:
1. Appendix A
2. Appendix B

CERTIFICATE OF COMPLIANCE NO. 1032

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR THE HI-STORM FW MPC STORAGE SYSTEM

TABLE OF CONTENTS

1.0 USE AND APPLICATION

---NOTE---
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

1.1 Definitions

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
DAMAGED FUEL ASSEMBLY	DAMAGED FUEL ASSEMBLIES are fuel assemblies with known or suspected cladding defects, as determined by a review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled 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 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 (DFC)	DFCs are specially designed enclosures for DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS which permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. DFCs authorized for use in the HI-STORM FW System are as follows: 1. Holtec Generic BWR design 2. Holtec Generic PWR design
FUEL DEBRIS	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.

1.1 Definitions

Term	Definition
FUEL BUILDING	The FUEL BUILDING is the site-specific power plant facility, governed by the regulations of 10 CFR Part 50, where the loaded OVERPACK or TRANSFER CASK is transferred to or from the transporter.
GROSSLY BREACHED SPENT FUEL ROD	*Spent nuclear fuel rod with a cladding defect that could lead to the release of fuel particulate greater than the average size fuel fragment for that particular assembly. A gross cladding breach may be confirmed by visual examination, through a review of reactor operating records indicating the presence of heavy metal isotopes, or other acceptable inspection means.*
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on an OVERPACK or TRANSFER CASK while it is being loaded with fuel assemblies. LOADING OPERATIONS begin when the first fuel assembly is placed in the MPC and end when the OVERPACK or TRANSFER CASK is suspended from or secured on the transporter. LOADING OPERATIONS does not include MPC TRANSFER.
MULTI-PURPOSE CANISTER (MPC)	MPCs are the sealed spent nuclear fuel canisters which consist of a honeycombed fuel basket contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. The MPC provides the confinement boundary for the contained radioactive materials.
MPC TRANSFER	MPC TRANSFER begins when the MPC is lifted off the TRANSFER CASK bottom lid and ends when the MPC is supported from beneath by the OVERPACK (or the reverse).
NON-FUEL HARDWARE	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

1.1 Definitions

Term	Definition
	assemblies, instrument tube tie rods (ITTRs), vibration suppressor inserts, and components of these devices such as individual rods.
OVERPACK	OVERPACKs are the casks which receive and contain the sealed MPCs for interim storage on the ISFSI. They provide gamma and neutron shielding, and provide for ventilated air flow to promote heat transfer from the MPC to the environs. The term OVERPACK does not include the TRANSFER CASK.
PLANAR-AVERAGE INITIAL ENRICHMENT	PLANAR AVERAGE INITIAL ENRICHMENT is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.
REPAIRED/RECONSITUTED FUEL ASSEMBLY	*Spent nuclear fuel assembly which contains dummy fuel rods that displaces an amount of water greater than or equal to the original fuel rods and/or which contains structural repairs so it can be handled by normal means.*
SPENT FUEL STORAGE CASKS (SFSCs)	SFSCs are containers approved for the storage of spent fuel assemblies at the ISFSI. The HI-STORM FW SFSC System consists of the OVERPACK and its integral MPC.
STORAGE OPERATIONS	STORAGE OPERATIONS include all licensed activities that are performed at the ISFSI while an SFSC containing spent fuel is situated within the ISFSI perimeter. STORAGE OPERATIONS does not include MPC TRANSFER.
TRANSFER CASK	TRANSFER CASKs are containers designed to contain the MPC during and after loading of spent fuel assemblies, and prior to and during unloading and to transfer the MPC to or from the OVERPACK.
TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all licensed activities performed on an OVERPACK or TRANSFER CASK loaded with one or more fuel assemblies when it is being moved after LOADING OPERATIONS or before UNLOADING OPERATIONS. TRANSPORT OPERATIONS begin when the OVERPACK or TRANSFER CASK is first suspended from or secured on the transporter and end when the OVERPACK or TRANSFER CASK is at its destination and no longer secured on or suspended from the transporter.

1.1 Definitions

Term	Definition
	TRANSPORT OPERATIONS includes MPC TRANSFER.
UNDAMAGED FUEL ASSEMBLY	UNDAMAGED FUEL ASSEMBLIES are*: a)* fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means*; or* ~~.~~*b)* ~~Fuel assemblies without fuel rods in fuel rod locations shall not be classified as UNDAMAGED 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).~~*a BWR fuel assembly with an intact channel, a maximum planar average initial of 3.3 wt% U-235, without known or suspected GROSSLY BREACHED SPENT FUEL RODS, and which can be handled by normal means. An UNDAMAGED FUEL ASSEMBLY may be a REPAIRED/RECONSTITUTED FUEL ASSEMBLY.*
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on an SFSC to be unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the OVERPACK or TRANSFER CASK is no longer suspended from or secured on the transporter and end when the last fuel assembly is removed from the SFSC. UNLOADING OPERATIONS does not include MPC TRANSFER.
ZR	ZR means any zirconium-based fuel cladding or fuel channel material authorized for use in a commercial nuclear power plant reactor.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE	The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.
BACKGROUND	Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors. When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

(continued)

1.2 Logical Connectors

EXAMPLES

The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 VERIFY . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Stop . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Remove . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three ACTIONS may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the HI-STORM FW System is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the HI-STORM FW System is not within the LCO Applicability. Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will not result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

(continued)

1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1	12 hours
	AND	
	B.2 Perform Action B.2	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to complete action B.1 within 12 hours AND complete action B.2 within 36 hours. A total of 12 hours is allowed for completing action B.1 and a total of 36 hours (not 48 hours) is allowed for completing action B.2 from the time that Condition B was entered. If action B.1 is completed within 6 hours, the time allowed for completing action B.2 is the next 30 hours because the total time allowed for completing action B.2 is 36 hours.

(continued)

Attachment 2 to Holtec letter 5018019
October 13, 2011

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One system not within limit.	A.1 Restore system to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1 Complete action B.1. AND B.2 Complete action B.2.	12 hours 36 hours

When a system is determined not to meet the LCO, Condition A is entered. If the system is not restored within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the system is restored after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

(continued)

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each component.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Restore compliance with LCO.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Complete action B.1. AND B.2 Complete action B.2.	6 hours 12 hours

The Note above the ACTIONS table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each component, and Completion Times tracked on a per component basis. When a component is determined to not meet the LCO, Condition A is entered and its Completion Time starts. If subsequent components are determined to not meet the LCO, Condition A is entered for each component and separate Completion Times start and are tracked for each component.

(continued)

1.3 Completion Times (continued)

IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

(continued)

1.4 Frequency (continued)

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify pressure within limit	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment or variables are outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 3.0.1.

If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a condition specified in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 3.0.4

(continued)

1.4 Frequency (continued)

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours prior to starting activity AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the example activity is to be performed, the Surveillance must be performed within 12 hours prior to starting the activity.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2.

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If the specified activity is canceled or not performed, the measurement of both intervals stops. New intervals start upon preparing to restart the specified activity.

2.0

This section is intentionally left blank

3.0 LIMITING CONDITIONS FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during specified conditions in the Applicability, except as provided in LCO 3.0.2.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.
LCO 3.0.3	Not applicable.
LCO 3.0.4	When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of an SFSC.
LCO 3.0.5	Equipment removed from service or not in service in compliance with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate it meets the LCO or that other equipment meets the LCO. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per..." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.3 (continued)	When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with Actions or that are related to the unloading of an SFSC.

3.1 SFSC INTEGRITY

3.1.1 Multi-Purpose Canister (MPC)

LCO 3.1.1 The MPC shall be dry and helium filled.

Table 3-1 provides decay heat and burnup limits for forced helium dehydration (FHD) and vacuum drying.

APPLICABILITY: Prior to TRANSPORT OPERATIONS

ACTIONS

---NOTES---

Separate Condition entry is allowed for each MPC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MPC cavity vacuum drying pressure or demoisturizer exit gas temperature limit not met.	A.1 Perform an engineering evaluation to determine the quantity of moisture left in the MPC.	7 days
	<u>AND</u>	
	A.2 Develop and initiate corrective actions necessary to return the MPC to compliance with Table 3-1.	30 days

ACTIONS (continued)

B. MPC helium backfill limit not met.	B.1 Perform an engineering evaluation to determine the impact of helium differential. AND B.2.1 Develop and initiate corrective actions necessary to return the MPC to an analyzed condition by adding helium to or removing helium from the MPC. OR B.2.2 Develop and initiate corrective actions necessary to demonstrate through analysis, using the models and methods from the HI-STORM FW FSAR, that all limits for MPC components and contents will be met.	72 hours 14 days
C. MPC helium leak rate limit for vent and drain port cover plate welds not met.	C.1 Perform an engineering evaluation to determine the impact of increased helium leak rate on heat removal capability and offsite dose. AND C.2 Develop and initiate corrective actions necessary to return the MPC to compliance with SR 3.1.1.3.	24 hours 7 days

D. Required Actions and associated Completion Times not met.	D.1 Remove all fuel assemblies from the SFSC.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify that the MPC cavity has been dried in accordance with the applicable limits in Table 3-1.	Once, prior to TRANSPORT OPERATIONS
SR 3.1.1.2	Verify MPC helium backfill quantity is within the limit specified in Table 3-2 for the applicable MPC model. Re-performance of this surveillance is not required upon successful completion of Action B.2.2.	Once, prior to TRANSPORT OPERATIONS
SR 3.1.1.3	Verify that the helium leak rate through the MPC vent port confinement weld meets the leaktight criteria of ANSI N14.5-1997 and verify that the helium leak rate through the MPC drain port confinement weld meets the leaktight criteria of ANSI N14.5-1997.	Once, prior to TRANSPORT OPERATIONS

3.1 SFSC INTEGRITY

3.1.2 SFSC Heat Removal System

LCO 3.1.2 The SFSC Heat Removal System shall be operable

---NOTE---
The SFSC Heat Removal System is operable when 50% or more of the inlet vent areas are unblocked and available for flow or when air temperature requirements are met.

APPLICABILITY: During STORAGE OPERATIONS.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each SFSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFSC Heat Removal System operable, but partially (<50%) blocked.	A.1 Remove blockage.	N/A
B. SFSC Heat Removal System inoperable.	B.1 Restore SFSC Heat Removal System to operable status.	8 hours
C. Required Action B.1 and associated Completion Time not met.	C.1 Measure SFSC dose rates in accordance with the Radiation Protection Program.	Immediately and once per 12 hours thereafter
	<u>AND</u>	
	C.2.1 Restore SFSC Heat Removal System to operable status.	24 hours
	<u>OR</u>	
	C.2.2 Transfer the MPC into a TRANSFER CASK.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2	Verify all OVERPACK inlets and outlets are free of blockage from solid debris or floodwater.	24 hours
	<u>OR</u> For OVERPACKS with installed temperature monitoring equipment, verify that the difference between the average OVERPACK air outlet temperature and ISFSI ambient temperature is ≤ 139°F for OVERPACKS containing PWR MPCs, ≤ 136°F for OVERPACKS containing BWR MPCs.	24 hours

3.1 SFSC INTEGRITY

3.1.3 MPC Cavity Reflooding

LCO 3.1.3 The MPC cavity pressure shall be < 100 psig

-----NOTE-----
The LCO is only applicable to wet UNLOADING OPERATIONS.

APPLICABILITY: UNLOADING OPERATIONS prior to and during re-flooding.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each MPC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MPC cavity pressure not within limit.	A.1 Stop re-flooding operations until MPC cavity pressure is within limit. AND	Immediately
	A.2 Ensure MPC vent port is not closed or blocked.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Ensure via analysis or direct measurement that MPC cavity pressure is within limit.	Once, prior to MPC re-flooding operations. OR Once every 1 hour thereafter when using direct measurement.

3.2 SFSC RADIATION PROTECTION.

3.2.1 TRANSFER CASK Surface Contamination.

LCO 3.2.1 Removable contamination on the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC shall each not exceed:

a. 1000 dpm/100 cm^2 from beta and gamma sources

b. 20 dpm/100 cm^2 from alpha sources.

-----NOTE-----
This LCO is not applicable to the TRANSFER CASK if MPC TRANSFER operations occur inside the FUEL BUILDING.

APPLICABILITY: During TRANSPORT OPERATIONS.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each TRANSFER CASK.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. TRANSFER CASK or MPC removable surface contamination limits not met.	A.1 Restore removable surface contamination to within limits.	7 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify that the removable contamination on the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC containing fuel is within limits.	Once, prior to TRANSPORT OPERATIONS

3.3 SFSC CRITICALITY CONTROL

3.3.1 Boron Concentration

LCO 3.3.1 The concentration of boron in the water in the MPC shall meet the following limits for the applicable MPC model and the most limiting fuel assembly array/class to be stored in the MPC:

MPC-37: Minimum soluble boron concentration as required by the table below†.

Array/Class	All Undamaged Fuel Assemblies		One or more Damaged Fuel Assemblies or Fuel Debris	
	Maximum Initial Enrichment ≤ 4.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment ≤ 4.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}U (ppmb)
All 14x14 and 16x16A	1000	1500	1300	1800
All 15x15 and 17x17	1500	2000	1800	2300

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

APPLICABILITY: During PWR fuel LOADING OPERATIONS with fuel and water in the MPC

AND

During PWR fuel UNLOADING OPERATIONS with fuel and water in the MPC.

ACTIONS

--NOTE--

Separate Condition entry is allowed for each MPC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend LOADING OPERATIONS or UNLOADING OPERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
-----NOTE----- This surveillance 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. -----		Once, within 4 hours prior to entering the Applicability of this LCO.
SR 3.3.1.1	Verify boron concentration is within the applicable limit using two independent measurements.	AND Once per 48 hours thereafter.

Table 3-1
MPC Cavity Drying Limits

Fuel Burnup (MWD/MTU)	MPC Heat Load (kW)	Method of Moisture Removal (Notes 1 and 2)
All Assemblies ≤ 45,000	≤ 47.05 (MPC-37) ≤ 46.36 (MPC-89)	VDS (Note 3) or FHD
One or more assemblies > 45,000	≤ 34.36 (MPC-37) ≤ 34.75 (MPC-89)	VDS (Notes 3 and 4) or FHD
One or more assemblies > 45,000	≤ 47.05 (MPC-37) ≤ 46.36 (MPC-89)	FHD

Notes:

1. VDS means a vacuum drying system. The acceptance criterion when using a VDS is the MPC cavity pressure shall be ≤ 3 torr for ≥ 30 minutes while the MPC is isolated from the vacuum pump.

2. FHD means a forced helium dehydration system. The acceptance criterion when using an FHD system is the gas temperature exiting the demoisturizer shall be ≤ 21°F for ≥ 30 minutes or the gas dew point exiting the MPC shall be ≤ 22.9°F for ≥ 30 minutes.

3. Vacuum drying of the MPC must be performed with the annular gap between the MPC and the TRANSFER CASK filled with water.

4. Heat load limits are set for each ~~Region~~*cell*; see Appendix B Table 2.3-3 for PWR fuel or Appendix B Table 2.3-4 for BWR.

Table 3-2
MPC Helium Backfill Limits[1]

~~MPC MODEL~~	~~LIMIT~~
~~MPC-37~~	
~~i. MPC Heat Load ≤ 37.6 kW~~ [2]	~~≥ 42.0 psig and ≤ 50.0 psig~~
~~ii. MPC Heat Load > 37.6 kW~~	~~≥ 42.5 psig and ≤ 45.5 psig~~
~~MPC-89~~	
~~i. MPC Heat Load ≤ 37.1 kW~~ [2]	~~≥ 42.0 psig and ≤ 50.0 psig~~
~~ii. MPC Heat Load > 37.1 kW~~	~~≥ 42.5 psig and ≤ 45.5 psig~~

MPC Model	*Decay Heat Limits Applied (per Appendix B Section 2.3)*	*% of Design Basis Heat Load*[2]	*Pressure range (psig)*
MPC-37	*Table 2.3-1, Pattern A*	*80*	*≥ 42.0 and ≤ 50.0*
MPC-37	*Table 2.3-1, Pattern A*	*90*	*≥ 42.0 and ≤ 47.8*
MPC-37	*Table 2.3-1, Pattern A*	*100*	*≥ 42.0 and ≤ 45.5*
MPC-37	*Table 2.3-1, Pattern B*	*100*	*≥ 41.0 and ≤ 46.0*
MPC-37	*Table 2.3-3*	*100*	*≥ 42.0 and ≤ 50.0*
MPC-89	*Table 2.3-2*	*80*	*≥ 42.0 and ≤ 50.0*
MPC-89	*Table 2.3-2*	*100*	*≥ 42.5 and ≤ 47.5*
MPC-89	*Table 2.3-4*	*100*	*≥ 42.0 and ≤ 50.0*

---

1 Helium used for backfill of MPC shall have a purity of ≥ 99.995%. Pressure range is at a reference temperature of 70°F

2 Percentage applies to each storage cell limit as listed in Appendix B Section 2.3 for the respective MPC and *heat load* ~~P~~pattern, *as applicable*.~~The individual storage cell heat loads must not exceed 80% of the design basis heat load listed in Appendix B Tables 2.3-1 and 2.3-2 for MPC-37 and MPC-89, respectively.~~

4.0

This section is intentionally left blank

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS

The following programs shall be established, implemented and maintained.

5.1 Radioactive Effluent Control Program

This program implements the requirements of 10 CFR 72.44(d).

a. The HI-STORM FW MPC Storage System does not create any radioactive materials or have any radioactive waste treatment systems. Therefore, specific operating procedures for the control of radioactive effluents are not required. Specification 3.1.1, Multi-Purpose Canister (MPC), provides assurance that there are not radioactive effluents from the SFSC.

b. This program includes an environmental monitoring program. Each general license user may incorporate SFSC operations into their environmental monitoring programs for 10 CFR Part 50 operations.

c. An annual report shall be submitted pursuant to 10 CFR 72.44(d)(3).

(continued)

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS (continued)

5.2 Transport Evaluation Program

a. For lifting of the loaded MPC, TRANSFER CASK, or OVERPACK using equipment which is integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply.

b. This program is not applicable when the TRANSFER CASK or OVERPACK is in the FUEL BUILDING or is being handled by equipment providing support from underneath (i.e., on a rail car, heavy haul trailer, air pads, etc...).

c. The TRANSFER CASK or OVERPACK, when loaded with spent fuel, may be lifted to and carried at any height necessary during TRANSPORT OPERATIONS and MPC TRANSFER, provided the lifting equipment is designed in accordance with items 1, 2, and 3 below.

   1. The metal body and any vertical columns of the lifting equipment shall be designed to comply with stress limits of ASME Section III, Subsection NF, Class 3 for linear structures. All vertical compression loaded primary members shall satisfy the buckling criteria of ASME Section III, Subsection NF.

   2. The horizontal cross beam and any lifting attachments used to connect the load to the lifting equipment shall be designed, fabricated, operated, tested, inspected, and maintained in accordance with applicable sections and guidance of NUREG-0612, Section 5.1. This includes applicable stress limits from ANSI N14.6.

   3. The lifting equipment shall have redundant drop protection features which prevent uncontrolled lowering of the load.

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS (continued)

5.3 Radiation Protection Program

5.3.1 Each cask user shall ensure that the Part 50 radiation protection program appropriately addresses dry storage cask loading and unloading, as well as ISFSI operations, including transport of the loaded OVERPACK or TRANSFER CASK outside of facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for direct radiation and contamination, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposures As Low As Reasonably Achievable (ALARA). The actions and criteria to be included in the program are provided below.

5.3.2 As part of its evaluation pursuant to 10 CFR 72.212(b)(2)(i)(C), the licensee shall perform an analysis to confirm that the dose limits of 10 CFR 72.104(a) will be satisfied under the actual site conditions and ISFSI configuration, considering the planned number of casks to be deployed and the cask contents.

5.3.3 Based on the analysis performed pursuant to Section 5.3.2, the licensee shall establish individual cask surface dose rate limits for the TRANSFER CASK and the OVERPACK to be used at the site. Total (neutron plus gamma) dose rate limits shall be established at the following locations:

a. The top of the OVERPACK.
b. The side OVERPACK
c. The side of the TRANSFER CASK
d. The inlet and outlet ducts on the OVERPACK

5.3.4 Notwithstanding the limits established in Section 5.3.3, the measured dose rates on a loaded OVERPACK or TRANSFER CASK shall not exceed the following values:

a. 30 mrem/hr (gamma + neutron) on the top of the OVERPACK
b. 300 mrem/hr (gamma + neutron) on the side of the OVERPACK, excluding inlet and outlet ducts
c. 3500 mrem/hr (gamma + neutron) on the side of the TRANSFER CASK

5.3.5 The licensee shall measure the TRANSFER CASK and OVERPACK surface neutron and gamma dose rates as described in Section 5.3.8 for comparison against the limits established in Section 5.3.3 or Section 5.3.4, whichever are lower.

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS (continued)

5.3 Radiation Protection Program (continued)

5.3.6 If the measured surface dose rates exceed the lower of the two limits established in Section 5.3.3 or Section 5.3.4, the licensee shall:

a. Administratively verify that the correct contents were loaded in the correct fuel storage cell locations.
b. Perform a written evaluation to verify whether an OVERPACK at the ISFSI containing the as-loaded MPC will cause the dose limits of 10 CFR 72.104 to be exceeded.
c. Perform a written evaluation within 30 days to determine why the surface dose rate limits were exceeded.

5.3.7 If the evaluation performed pursuant to Section 5.3.6 shows that the dose limits of 10 CFR 72.104 will be exceeded, the OVERPACK shall not be moved to the ISFSI or, in the case of the OVERPACK loaded at the ISFSI, the MPC shall be removed from the ISFSI until appropriate corrective action is taken to ensure the dose limits are not exceeded.

5.3.8 TRANSFER CASK and OVERPACK surface dose rates shall be measured at approximately the following locations:

a. A dose rate measurement shall be taken on the top of the OVERPACK at approximately the center of the lid.

b. A minimum of twelve (12) dose rate measurements shall be taken on the side of the OVERPACK in three sets of four measurements. One measurement set shall be taken approximately at the cask mid-height plane, 90 degrees apart around the circumference of the cask. The second and third measurement sets shall be taken approximately 60 inches above and below the mid-height plane, respectively, also 90 degrees apart around the circumference of the cask.

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS (continued)

5.3 Radiation Protection Program (continued)

c. A minimum of four (4) dose rate measurements shall be taken on the side of the TRANSFER CASK approximately at the cask mid-height plane. The measurement locations shall be approximately 90 degrees apart around the circumference of the cask. Dose rates shall be measured between the radial ribs of the water jacket.

d. A dose rate measurement shall be taken on contact at the surface of each inlet and outlet vent duct screen of the OVERPACK.

CERTIFICATE OF COMPLIANCE NO. 1032

APPENDIX B

APPROVED CONTENTS AND DESIGN FEATURES

FOR THE HI-STORM FW MPC STORAGE SYSTEM

TABLE OF CONTENTS

1.0 Definitions

Refer to Appendix A for Definitions.

2.0 APPROVED CONTENTS

2.1 Fuel Specifications and Loading Conditions

2.1.1 <u>Fuel to Be Stored in the HI-STORM FW MPC Storage System</u>

a. UNDAMAGED FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, FUEL DEBRIS, and NON-FUEL HARDWARE meeting the limits specified in Table 2.1-1 and other referenced tables may be stored in the HI-STORM FW MPC Storage System.

b. All BWR fuel assemblies may be stored with or without ZR channels.

2.1.2 <u>Fuel Loading</u>

Figures 2.1-1 and 2.1-2 define the regions for the MPC-37 and MPC-89 models, respectively. Fuel assembly decay heat limits are specified in Section 2.3.1. Fuel assemblies shall meet all other applicable limits specified in Tables 2.1-1 through 2.1-3.

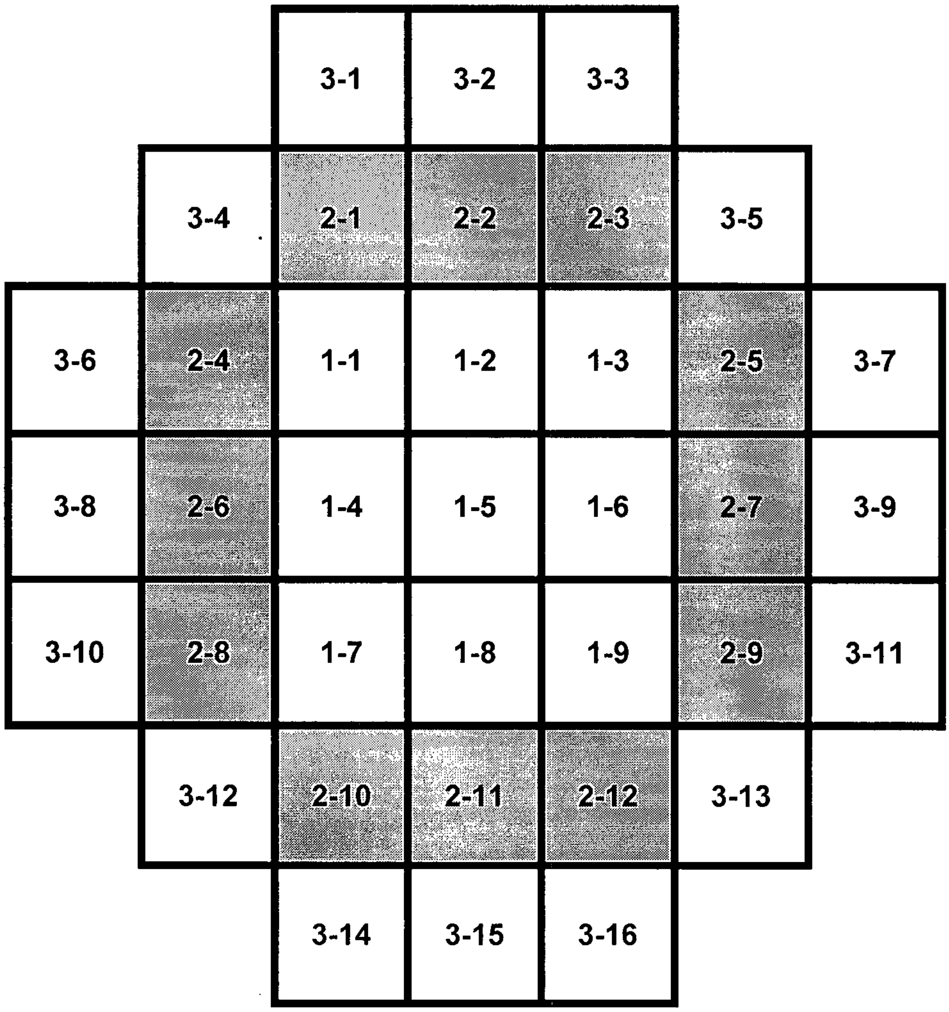
2.2 Violations

If any Fuel Specifications or Loading Conditions of 2.1 are violated, the following actions shall be completed:

2.2.1 The affected fuel assemblies shall be placed in a safe condition.

2.2.2 Within 24 hours, notify the NRC Operations Center.

2.2.3 Within 30 days, submit a special report which describes the cause of the violation, and actions taken to restore compliance and prevent recurrence.



Legend

Region-Cell ID

Figure 2.1-1
MPC-37 Region-Cell Identification

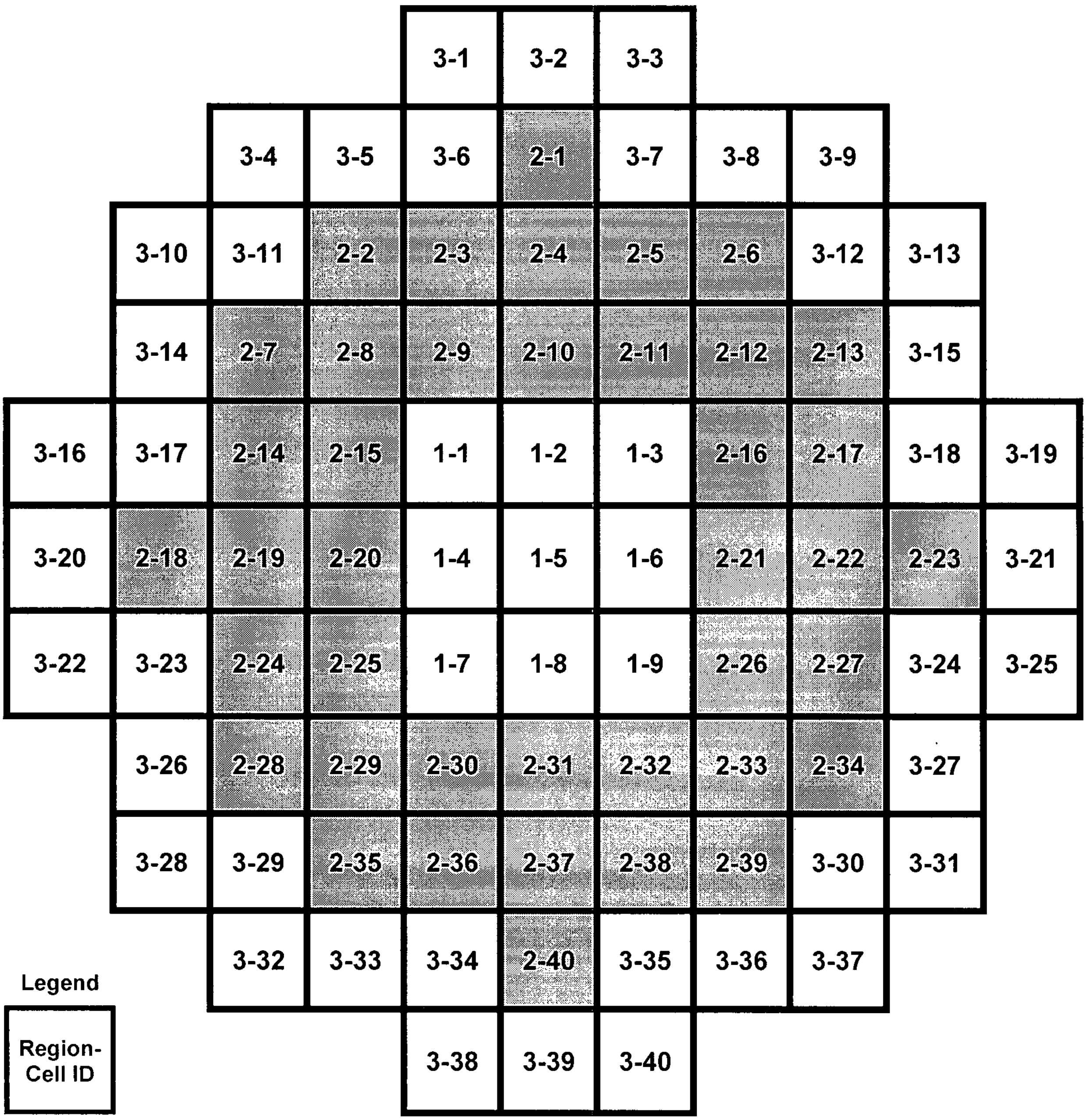


Figure 2.1-2
MPC-89 Region-Cell Identification

Table 2.1-1 (page 1 of 4)
Fuel Assembly Limits

I. MPC MODEL: MPC-37

A. Allowable Contents

1. Uranium oxide PWR UNDAMAGED FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and/or FUEL DEBRIS meeting the criteria in Table 2.1-2, with or without NON-FUEL HARDWARE and meeting the following specifications (Note 1):

a. Cladding Type:	ZR
b. Maximum Initial Enrichment:	5.0 wt. % U-235 with soluble boron credit per LCO 3.3.1
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling Time ≥ 3 years Assembly Average Burnup ≤ 68.2 GWD/MTU
d. Decay Heat Per Fuel Storage Location:	As specified in Section 2.3
e. Fuel Assembly Length:	≤ 199.2 inches (nominal design including NON-FUEL HARDWARE and DFC)
f. Fuel Assembly Width:	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight:	≤ 2050 lbs (including NON-FUEL HARDWARE and DFC)

Table 2.1-1 (page 2 of 4)
Fuel Assembly Limits

I. MPC MODEL: MPC-37 (continued)

B. Quantity per MPC: 37 FUEL ASSEMBLIES with up to twelve (12) DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS in DAMAGED FUEL CONTAINERS (DFCs). DFCs may be stored in fuel storage locations 3-1, 3-3 through 3-7, 3-10 through 3-14, and 3-16 (see Figure 2.1-1). The remaining fuel storage locations may be filled with PWR UNDAMAGED FUEL ASSEMBLIES meeting the applicable specifications.

C. One (1) Neutron Source Assembly (NSA) is authorized for loading in the MPC-37.

D. Up to thirty (30) BRPAs are authorized for loading in the MPC-37.

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, or vibration suppressor inserts, with or without ITTRs, may be stored in any fuel storage location. Fuel assemblies containing APSRs, RCCAs, CEAs, CRAs, or NSAs may only be loaded in fuel storage Regions 1 and 2 (see Figure 2.1-1).

Table 2.1-1 (page 3 of 4)
Fuel Assembly Limits

II. MPC MODEL: MPC-89

A. Allowable Contents

1. Uranium oxide BWR UNDAMAGED FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and/or FUEL DEBRIS meeting the criteria in Table 2.1-3, with or without channels and meeting the following specifications:

a. Cladding Type:	ZR
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT(Note 1):	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment	5.0 wt. % U-235
d. Post-irradiation Cooling Time and Average Burnup Per Assembly	
i. Array/Class 8x8F	Cooling time ≥ 10 years and an assembly average burnup ≤ 27.5 GWD/MTU.
ii. All Other Array Classes	Cooling Time ≥ 3 years and an assembly average burnup ≤ 65 GWD/MTU
e. Decay Heat Per Assembly	
i. Array/Class 8x8F	≤ 183.5 Watts
ii. All Other Array Classes	As specified in Section 2.3
f. Fuel Assembly Length	≤ 176.5 inches (nominal design)
g. Fuel Assembly Width	≤ 5.95 inches (nominal design)
h. Fuel Assembly Weight	≤ 850 lbs, including a DFC as well as a channel

Table 2.1-1 (page 4 of 4)
Fuel Assembly Limits

II. MPC MODEL: MPC-89 (continued)

B. Quantity per MPC: 89 FUEL ASSEMBLIES with up to sixteen (16) DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS in DAMAGED FUEL CONTAINERS (DFCs). DFCs may be stored in fuel storage locations 3-1, 3-3, 3-4, 3-9, 3-10, 3-13, 3-16, 3-19, 3-22, 3-25, 3-28, 3-31, 3-32, 3-37, 3-38, and 3-40 (see Figure 2.1-2). The remaining fuel storage locations may be filled with BWR UNDAMAGED FUEL ASSEMBLIES meeting the applicable specifications.

Note 1: The lowest maximum allowable enrichment of any fuel assembly loaded in an MPC-89, based on fuel array class and fuel classification, is the maximum allowable enrichment for the remainder of the assemblies loaded in that MPC.

Table 2.1-2 (page 1 of 3) 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.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.420	≥ 0.417
Fuel Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3736	≤ 0.3640
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3671	≤ 0.3570
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.563	≤ 0.563
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	17	17	5 (Note 2)	21	21
Guide/Instrument Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.015	≥ 0.0165

Table 2.1-2 (page 2 of 3)
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.)	≥ 0.430	≥ 0.428	≥ 0.428	≥ 0.414	≥ 0.413
Fuel Clad I.D. (in.)	≤ 0.3800	≤ 0.3790	≤ 0.3820	≤ 0.3700	≤ 0.3670
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3735	≤ 0.3707	≤ 0.3742	≤ 0.3622	≤ 0.3600
Fuel Rod Pitch (in.)	≤ 0.568	≤ 0.568	≤ 0.568	≤ 0.568	≤ 0.550
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	17	17	17	17	9 (Note 4)
Guide/Instrument Tube Thickness (in.)	≥ 0.0150	≥ 0.0140	≥ 0.0140	≥ 0.0140	≥ 0.0140

Table 2.1-2 (page 3 of 3)
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.)	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377	≥ 0.372	≥ 0.372
Fuel Clad I.D. (in.)	≤ 0.3350	≤ 0.3150	≤ 0.3310	≤ 0.3330	≤ 0.3310	≤ 0.3310
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252	≤ 0.3232	≤ 0.3232
Fuel Rod Pitch (in.)	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502	≤ 0.496	≤ 0.496
Active Fuel length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 170	≤ 170
No. of Guide and/or Instrument Tubes	5 (Note 2)	25	25	25	25	24
Guide/Instrument Tube Thickness (in.)	≥ 0.0350	≥ 0.016	≥ 0.014	≥ 0.020	≥ 0.014	≥ 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)

Attachment 2 to Holtec letter 5018019
October 13, 2011

Table 2.1-3 (page 1 of 4) 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.% ^{235}U) *(Note 14)*	≤ 4.8	≤ 4.8	≤ 4.8	≤ 4.8	≤ 4.8
No. of Fuel Rod Locations (Full Length or Total/Full Length)	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 (2 of 4)
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.% ^{235}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 (page 3 of 4) BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)					
Fuel Assembly Array and Class	**9x9 E (Note 2)**	**9x9 F (Note 2)**	**9x9 G**	**10x10 A**	**10x10 B**
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U) *(Note 14)*	$\leq$ 4.5 (Note 12)	$\leq$ 4.5 (Note 12)	$\leq$ 4.8	$\leq$ 4.8	$\leq$ 4.8
No. of Fuel Rod Locations	76	76	72	92/78 (Note 7)	91/83 (Note 8)
Fuel Clad O.D. (in.)	$\geq$0.4170	$\geq$0.4430	$\geq$0.4240	$\geq$0.4040	$\geq$0.3957
Fuel Clad I.D. (in.)	$\leq$0.3640	$\leq$0.3860	$\leq$0.3640	$\leq$ 0.3520	$\leq$ 0.3480
Fuel Pellet Dia. (in.)	$\leq$0.3530	$\leq$0.3745	$\leq$0.3565	$\leq$ 0.3455	$\leq$ 0.3420
Fuel Rod Pitch (in.)	$\leq$ 0.572	$\leq$ 0.572	$\leq$ 0.572	$\leq$ 0.510	$\leq$ 0.510
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Water Rods (Note 10)	5	5	1 (Note 5)	2	1 (Note 5)
Water Rod Thickness (in.)	$\geq$0.0120	$\geq$0.0120	$\geq$0.0320	$\geq$0.0300	> 0.00
Channel Thickness (in.)	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120

Table 2.1-3 (page 4 of 4) BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)			
Fuel Assembly Array and Class	**10x10 C**	**10x10 F**	**10x10 G**
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U) *(Note 14)*	$\leq$ 4.8	$\leq$ 4.7 (Note 13)	$\leq$ 4.6 (Note 12)
No. of Fuel Rod Locations	96	92/78 (Note 7)	96/84
Fuel Clad O.D. (in.)	$\geq$ 0.3780	$\geq$ 0.4035	$\geq$ 0.387
Fuel Clad I.D. (in.)	$\leq$ 0.3294	$\leq$ 0.3570	$\leq$ 0.340
Fuel Pellet Dia. (in.)	$\leq$ 0.3224	$\leq$ 0.3500	$\leq$ 0.334
Fuel Rod Pitch (in.)	$\leq$ 0.488	$\leq$ 0.510	$\leq$ 0.512
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Water Rods (Note 10)	5 (Note 9)	2	5 (Note 9)
Water Rod Thickness (in.)	$\geq$ 0.031	$\geq$ 0.030	$\geq$ 0.031
Channel Thickness (in.)	$\leq$ 0.055	$\leq$ 0.120	$\leq$ 0.060

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 f*~~F~~uel assemblies classified as DAMAGED FUEL*, all assemblies in the MPC* are limited to 4.0 wt.% U-235.
13. *When loading* ~~F~~*f*uel assemblies classified as DAMAGED FUEL*, all assemblies in the MPC* are limited to 4.6 wt.% U-235.
14. *In accordance with the definition of UNDAMAGED FUEL, 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.*

2.3 Decay Heat Limits

This section provides the limits on fuel assembly decay heat for storage in the HI-STORM FW System. The method to verify compliance, including examples, is provided in Chapter 13 of the HI-STORM FW FSAR.

2.3.1 Fuel Loading Decay Heat Limits

Table 2.3-1 provides the maximum allowable decay heat per fuel storage location for MPC-37. Table 2.3-2 provides the maximum allowable decay heat per fuel storage location for MPC-89. The limits in these tables are applicable when using FHD to dry moderate or high burnup fuel and when using VDS to dry moderate burnup fuel only. Tables 2.3-3 and 2.3-4 provide the maximum allowable decay heat per fuel storage location for MPC-37 and MPC-89, respectively, when using VDS to dry high burnup fuel.

TABLE 2.3-1
MPC-37 HEAT LOAD DATA (See Figure 2.1-1)

Number of Regions: 3

Number of Storage Cells: 37

Maximum Heat Load *(kW)*: 47.05 *(Pattern A); 46.8 (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	1.13	*1.2*	9	10.17	*10.8*
2	1.78	*1.2*	12	21.36	*14.4*
3	0.97	*1.35*	16	15.52	*21.6*

TABLE 2.3-2
MPC-89 HEAT LOAD DATA (See Figure 2.1-2)

Number of Regions: 3

Number of Storage Cells: 89

Maximum 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

TABLE 2.3-3 MPC-37 HEAT LOAD DATA (See Figure 2.1-1)			
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

TABLE 2.3-4 MPC-89 HEAT LOAD DATA (See Figure 2.1-2)			
Number of Regions: 3			
Number of Storage Cells: 89			
Maximum Heat Load: 34.75 kW			
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

2.3.2 When complying with the maximum fuel storage location decay heat limits, users must account for the decay heat from both the fuel assembly and any NON-FUEL HARDWARE, as applicable for the particular fuel storage location, to ensure the decay heat emitted by all contents in a storage location does not exceed the limit.

3.0 DESIGN FEATURES

3.1 Site

3.1.1 Site Location

The HI-STORM FW Cask System is authorized for general use by 10 CFR Part 50 license holders at various site locations under the provisions of 10 CFR 72, Subpart K.

3.2 Design Features Important for Criticality Control

3.2.1 MPC-37

1. Basket cell ID: 8.92 in. (min.)
2. Basket cell wall thickness: 0.57 in. (min.)
3. B_4C in the Metamic-HT: 10.0 wt % (min.)

3.2.2 MPC-89

1. Basket cell ID: 5.99 in. (min.)
2. Basket cell wall thickness: 0.38 in. (min.)
3. B_4C in the Metamic-HT: 10.0 wt % (min.)

3.2.3 Neutron Absorber Tests

Section 10.1.6.3 of the HI-STORM FW FSAR is hereby incorporated by reference into the HI-STORM FW CoC.

3.3 Codes and Standards

The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 2007 Edition, is the governing Code for the HI-STORM FW System MPC as clarified in Specification 3.3.1 below, except for Code Sections V and IX. The ASME Code paragraphs applicable to the HI-STORM FW OVERPACK and TRANSFER CASK are listed in Table 3-2. The latest effective editions of ASME Code Sections V and IX, including addenda, may be used for activities governed by those sections, provided a written reconciliation of the later edition against the 2007 Edition, including any addenda, is performed by the certificate holder. American Concrete Institute (ACI) 349-85 is the governing Code for plain concrete as clarified in Appendix 1.D of the Final Safety Analysis Report for the HI-STORM 100 Cask System.

3.3.1 <u>Alternatives to Codes, Standards, and Criteria</u>

Table 3-1 lists approved alternatives to the ASME Code for the design of the MPCs of the HI-STORM FW Cask System.

3.3.2 <u>Construction/Fabrication Alternatives to Codes, Standards, and Criteria</u>

Proposed alternatives to the ASME Code, Section III, 2007 Edition, including modifications to the alternatives allowed by Specification 3.3.1 may be used on a case-specific basis when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternative should demonstrate that:

1. The proposed alternatives would provide an acceptable level of quality and safety, or
2. Compliance with the specified requirements of the ASME Code, Section III, 2007 Edition, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Requests for alternatives shall be submitted in accordance with 10 CFR 72.4.

(continued)

3.0 DESIGN FEATURES (continued)

TABLE 3-1
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.	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. 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.
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.

TABLE 3-1
List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)

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 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.	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.
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.

TABLE 3-1
List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)

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.
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.

Attachment 2 to Holtec letter 5018019
October 13, 2011

TABLE 3-1
List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)

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 welded in the field following fuel assembly loading. After the lid to shell weld is completed, the MPC 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. 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. The inspection results, including relevant findings (indications) shall be made a permanent part of the user's records by video, photographic, of 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.

TABLE 3-1
List of ASME Code Alternatives for Multi-Purpose Canisters (MPCs)

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.

TABLE 3-2 REFERENCE ASME CODE PARAGRAPHS FOR HI-STORM FW OVERPACK and HI-TRAC VW TRANSFER CASK, PRIMARY LOAD BEARING PARTS			
	Item	**Code Paragraph†**	**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

† All references to the ASME Code refer to applicable sections of the 2007 edition.

3.0 DESIGN FEATURES (continued)

3.4 Site-Specific Parameters and Analyses

Site-specific parameters and analyses that will require verification by the system user are, as a minimum, as follows:

1. The temperature of 80° F is the maximum average yearly temperature.

2. The allowed temperature extremes, averaged over a 3-day period, shall be greater than -40° F and less than 125° F.

3. a. The resultant horizontal acceleration (vectorial sum of two horizontal Zero Period Accelerations (ZPAs) at a three-dimensional seismic site), a_H, and vertical ZPA, a_V, on the top surface of the ISFSI pad, expressed as fractions of a, shall satisfy the following inequalities:

$$a_H \leq f \,\square (1 - a_V); \text{ and}$$

$$a_H \leq r\,(1 - a_V) / h$$

where f is the Coulomb friction coefficient for the cask/ISFSI pad interface, r is the radius of the cask, and h is the height of the cask center-of-gravity above the ISFSI pad surface. Unless demonstrated by appropriate testing that a higher coefficient of friction value is appropriate for a specific ISFSI, the value used shall be 0.53. If acceleration time-histories on the ISFSI pad surface are available, a_H and a_V may be the coincident values of the instantaneous net horizontal and vertical accelerations. If instantaneous accelerations are used, the inequalities shall be evaluated at each time step in the acceleration time history over the total duration of the seismic event.

If this static equilibrium based inequality cannot be met, a dynamic analysis of the cask/ISFSI pad assemblage with appropriate recognition of soil/structure interaction effects shall be performed to ensure that the casks will not tip over or undergo excessive sliding under the site's Design Basis Earthquake.

b. Under environmental conditions that may degrade the pad/cask interface friction (such as due to icing) the response of the casks under the site's Design Basis Earthquake shall be established using the best estimate of the friction coefficient in an appropriate analysis model. The analysis should demonstrate that the earthquake will not result in cask tipover or cause excessive sliding such that impact between casks could occur. Any impact between casks should be considered an accident for which the maximum total deflection, d, in the active fuel region of the basket panels shall be limited by the following inequality: $d \leq 0.005\ l$, where l is the basket cell inside dimension.

4. The maximum permitted depth of submergence under water shall not exceed 125 feet.

5. The maximum permissible velocity of floodwater, V, for a flood of height, h, shall be the lesser of V_1 or V_2, where:

$$V_1 = (1.876\ W^*)^{1/2} / h$$

$$V_2 = (1.876\ f\ W^* / D\ h)^{1/2}$$

and W* is the apparent (buoyant weight) of the loaded overpack (in pounds force), D is the diameter of the overpack (in feet), and f □is the interface coefficient of friction between the ISFSI pad and the overpack, as used in step 3.a above. Use the height of the overpack, H, if h>H.

6. The potential for fire and explosion while handling a loaded OVERPACK or TRANSFER CASK shall be addressed, based on site-specific considerations. The user shall demonstrate that the site-specific potential for fire is bounded by the fire conditions analyzed by the Certificate Holder, or an analysis of the site-specific fire considerations shall be performed.

7. The user shall demonstrate that the ISFSI pad parameters used in the non-mechanistic tipover analysis are bounding for the site or a site specific non-mechanistic tipover analysis shall be performed using the dynamic model described in FSAR Section 3.4. The maximum total deflection, d, in the active fuel region of the basket panels shall be limited by the following inequality: $d \leq 0.005\ \ell$, where ℓ is basket cell inside dimension.

8. In cases where engineered features (i.e., berms and shield walls) are used to ensure that the requirements of 10CFR72.104(a) are met, such features are to be considered important-to-safety and must be evaluated to determine the applicable quality assurance category.

9. LOADING OPERATIONS, TRANSPORT OPERATIONS, and UNLOADING OPERATIONS shall only be conducted with working area ambient temperatures $\geq 0^\circ$ F.

10. For those users whose site-specific design basis includes an event or

events (e.g., flood) that result in the blockage of any OVERPACK inlet or outlet air ducts for an extended period of time (i.e, longer than the total Completion Time of LCO 3.1.2), an analysis or evaluation may be performed to demonstrate adequate heat removal is available for the duration of the event. Adequate heat removal is defined as fuel cladding temperatures remaining below the short term temperature limit. If the analysis or evaluation is not performed, or if fuel cladding temperature limits are unable to be demonstrated by analysis or evaluation to remain below the short term temperature limit for the duration of the event, provisions shall be established to provide alternate means of cooling to accomplish this objective.

11. Users shall establish procedural and/or mechanical barriers to ensure that during LOADING OPERATIONS and UNLOADING OPERATIONS, either the fuel cladding is covered by water, or the MPC is filled with an inert gas.

12. The entire haul route shall be evaluated to ensure that the route can support the weight of the loaded system and its conveyance.

13. The loaded system and its conveyance shall be evaluated to ensure under the site specific Design Basis Earthquake the system does not tipover or slide off the haul route.

14. The HI-STORM FW/HI-TRAC VW stack which occurs during MPC TRANSFER shall be evaluated to ensure under the site specific Design Basis Earthquake the system does not tipover. A probabilistic risk assessment cannot be used to rule out the occurrence of the earthquake during MPC TRANSFER.

(continued)

3.0 DESIGN FEATURES (continued)

3.5 Combustible Gas Monitoring During MPC Lid Welding and Cutting

During MPC lid-to-shell welding and cutting operations, combustible gas monitoring of the space under the MPC lid is required, to ensure that there is no combustible mixture present.

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 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.

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.

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. A small axial clearance is provided to account for 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 Tables 1.2.7 and 1.2.8. 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^{o} 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.

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.

Finally, the HI-STORM FW overpack features heat shields engineered to protect the overpack body concrete and the overpack lid concrete from excessive temperature rise due to radiant heat from the MPC. A thin cylindrical steel liner, concentric with the inner shell of the overpack, but slightly smaller in diameter, hangs from the top array of MPC guides. A separate thin steel liner is welded to the underside of the overpack lid. The heat shields are depicted in the licensing drawings in Section 1.5.

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.

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 top forging of the transfer cask using a set of high strength bolts. A 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 **V**ariable **W**eight 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 in Appendix 1.B.

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 four 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.

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.

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 aggregate. Users of HI-STORM 100 systems have used concrete of density approaching 200 lb/ft^3 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 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.

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†:	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†:	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.

† 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†/-40††	752†/-40††
Design internal pressure (psig) Normal conditions Off-normal conditions Accident Conditions	 100 120 200	 100 120 200
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) Short Term Operations (°F) Off-normal and Accident (°F)	 752 752 or 1058††† 1058	 752 752 or 1058††† 1058
Maximum permissible multiplication factor (k_{eff}) including all uncertainties and biases	< 0.95	< 0.95
B_4C content (by weight) (min.) in the Metamic-HT Neutron Absorber (storage cell walls)	10%	10%
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

† 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.

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

††† 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*(a)* *Pattern A* MPC-37 HEAT LOAD DATA (See Figure 1.2.1)			
Number of Regions: 3			
Number of Storage Cells: 37			
Maximum Heat Load: 47.05 *kW*			
Region No.	Decay Heat Limit per Cell, kW	Number of Cells per Region	Decay Heat Limit per Region, kW
1	1.13	9	10.17
2	1.78	12	21.36
3	0.97	16	15.52

TABLE 1.2.3(b) *Pattern B* *MPC-37 HEAT LOAD DATA (See Figure 1.2.1)*			
Number of Regions: 3			
Number of Storage Cells: 37			
Maximum Heat Load: 46.8 kW			
Region No.	*Decay Heat Limit per Cell, kW*	*Number of Cells per Region*	*Decay Heat Limit per Region, kW*
1	*1.2*	*9*	*10.8*
2	*1.2*	*12*	*14.4*
3	*1.35*	*16*	*21.6*

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

TABLE 1.2.4 MPC-89 HEAT LOAD DATA (See Figure 1.2.2)			
Number of Regions: 3			
Number of Storage Cells: 89			
Maximum 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		
Item	**Property**	**Value**
Metamic-HT Neutron Absorber	Nominal Thickness (mm)	10 (MPC-89) 15 (MPC-37)
	Minimum B_4C Weight %	10 (MPC-89) 10 (MPC-37)
Concrete in HI-STORM FW overpack body and lid	Installed Nominal Density (lb/ft^3)	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**†	**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

† 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

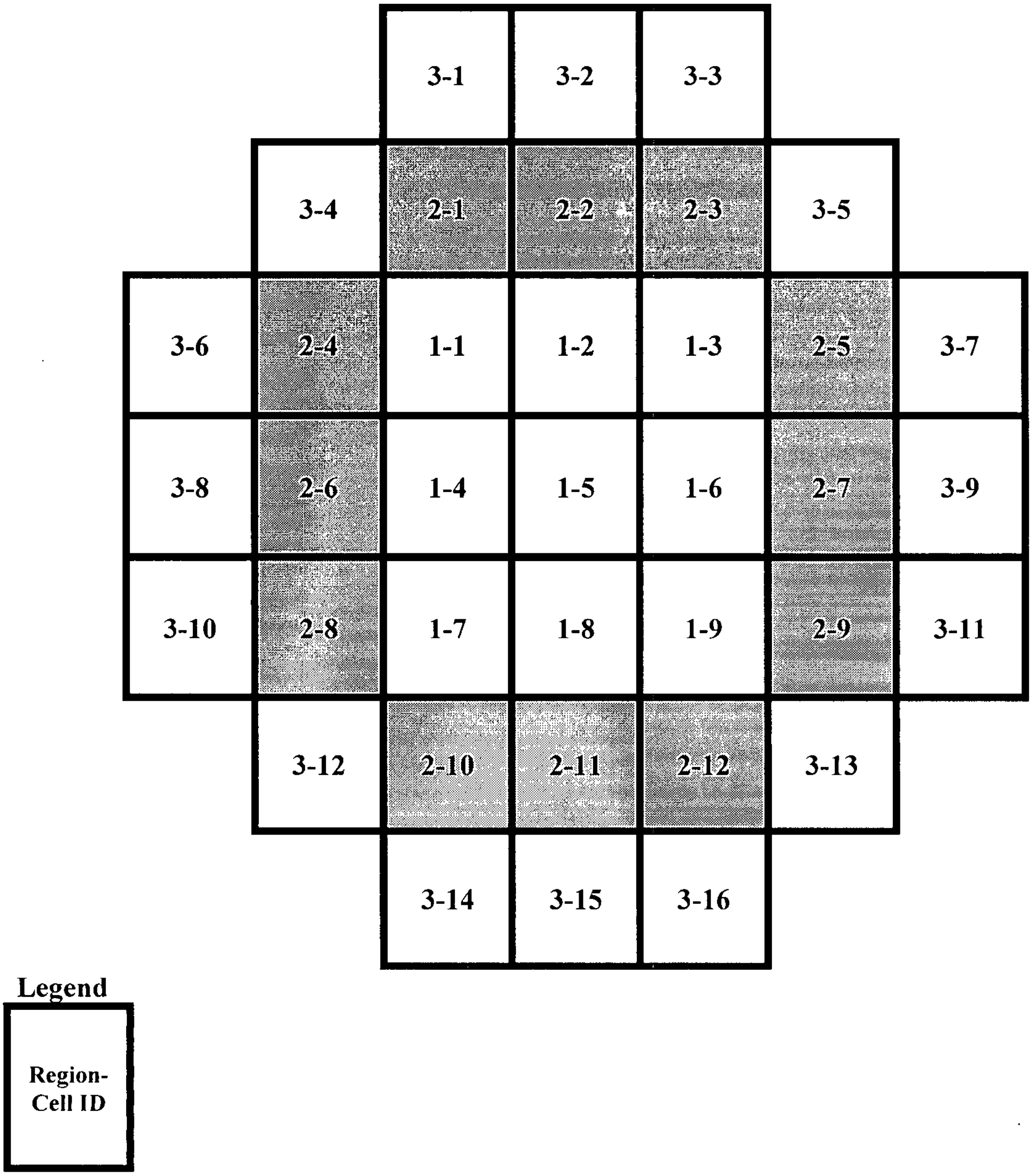


Figure 1.2.1: MPC-37 Basket, Region and Cell Identification

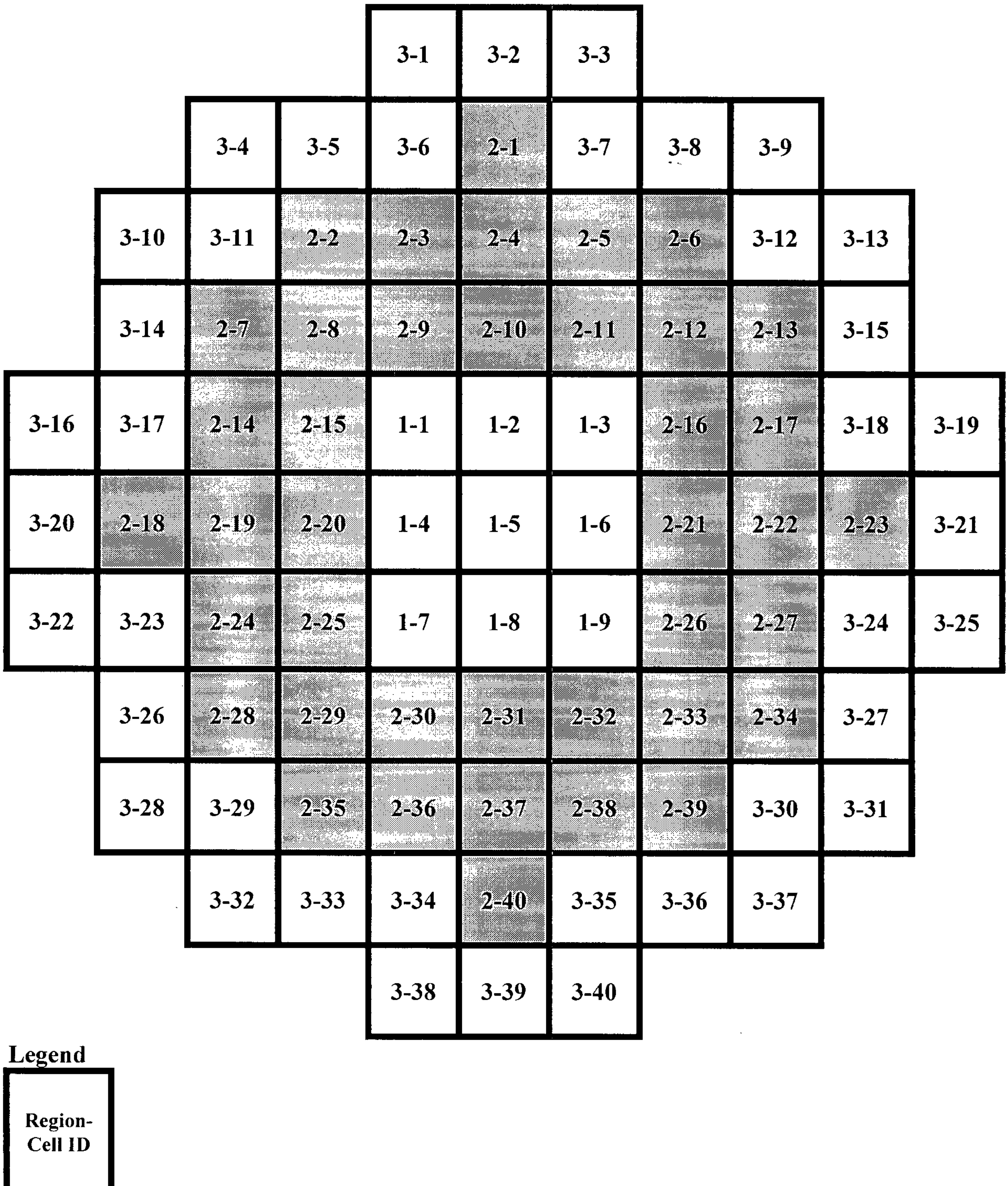


Figure 1.2.2: MPC-89 Basket, Region and Cell Identification

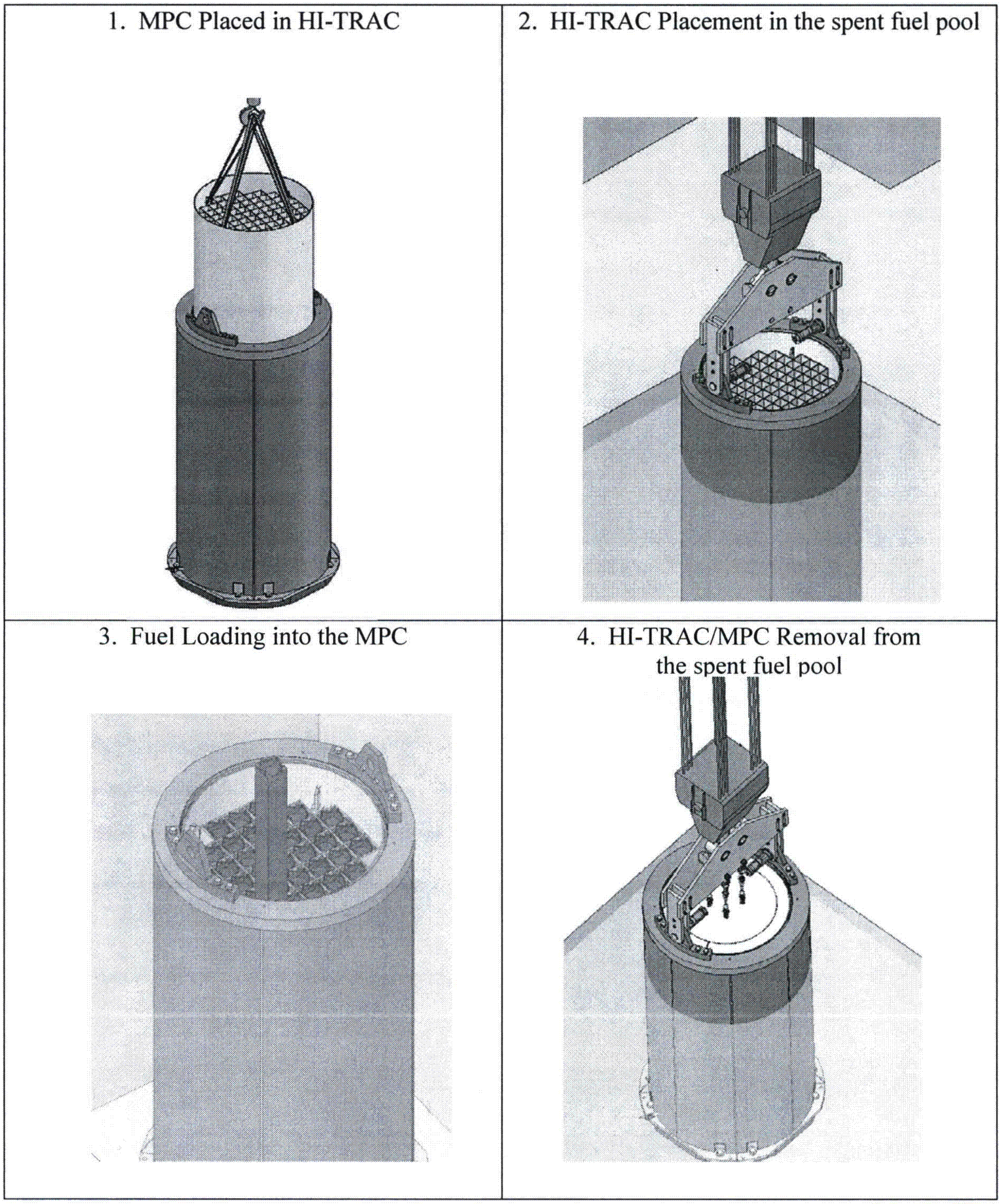


FIGURE 1.2.3: SUMMARY OF TYPICAL LOADING OPERATIONS

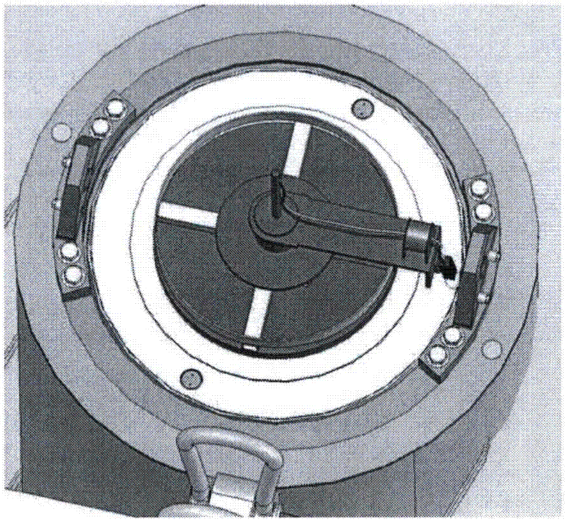
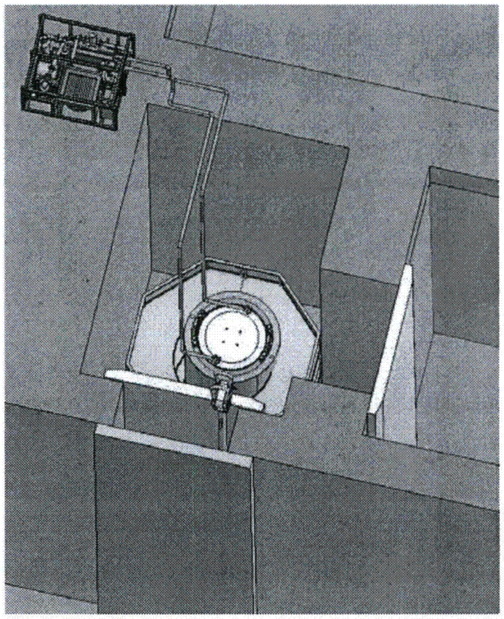
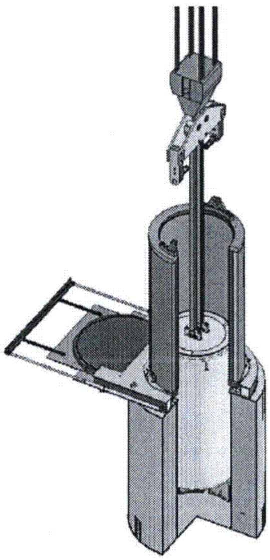
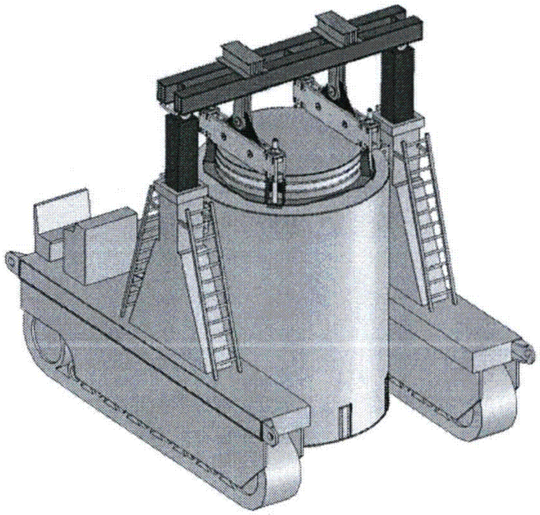
5. MPC Closure Operations (Lid to Shell Welding) 	6. MPC Draining, Drying and Backfill 
7. System Stackup and MPC Transfer Operations 	8. HI-STORM Movement to the ISFSI 

FIGURE 1.2.3 (CONTINUED): SUMMARY OF TYPICAL LOADING OPERATIONS

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 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 (Note 1)**	
	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	≤ 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†: - 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

† 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.

Table 2.1.1

MATERIAL TO BE STORED

PARAMETER	VALUE (Note 1)	
	MPC-37	MPC-89
Fuel Assembly Nominal Length (in.)	Minimum: 157 (with NFH) Reference: 167.2 (with NFH) Maximum: 199.2 (with NFH and DFC)	Minimum: 171 Reference: 176.5 Maximum: 176.5 (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)
Other Limitations	▪ 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. ▪ One NSA. ▪ Up to 30 BPRAs. ▪ 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. ▪ CRAs, RCCAs, CEAs, NSAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations specified in Figure 2.1.5.	▪ 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.

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	$\leq$ 0.3734	$\leq$ 0.3880	$\leq$ 0.3736	$\leq$ 0.3640
Fuel Pellet Dia. (in.) (Note 3)	$\leq$ 0.3444	$\leq$ 0.3659	$\leq$ 0.3805	$\leq$ 0.3671	$\leq$ 0.3570
Fuel Rod Pitch (in.)	$\leq$ 0.556	$\leq$ 0.556	$\leq$ 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.)	≥ 0.430	≥ 0.428	≥ 0.428	≥ 0.414	≥ 0.413
Fuel Clad I.D. (in.)	≤ 0.3800	≤ 0.3790	≤ 0.3820	≤ 0.3700	≤ 0.3670
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3735	≤ 0.3707	≤ 0.3742	≤ 0.3622	≤ 0.3600
Fuel Rod Pitch (in.)	≤ 0.568	≤ 0.568	≤ 0.568	≤ 0.568	≤ 0.550
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	17	17	17	17	9 (Note 4)
Guide/Instrument Tube Thickness (in.)	≥ 0.0150	≥ 0.0140	≥ 0.0140	≥ 0.0140	≥ 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.)	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377	≥ 0.372	≥ 0.372
Fuel Clad I.D. (in.)	≤ 0.3350	≤ 0.3150	≤ 0.3310	≤ 0.3330	≤ 0.3310	≤ 0.3310
Fuel Pellet Dia. (in.) (Note 3)	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252	≤ 0.3232	≤ 0.3232
Fuel Rod Pitch (in.)	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502	≤ 0.496	≤ 0.496
Active Fuel length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 170	≤ 170
No. of Guide and/or Instrument Tubes	5 (Note 2)	25	25	25	25	24
Guide/Instrument Tube Thickness (in.)	≥ 0.0350	≥ 0.016	≥ 0.014	≥ 0.020	≥ 0.014	≥ 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.% ^{235}U) *(Note 14)*	$\leq$ 4.8	$\leq$ 4.8	$\leq$ 4.8	$\leq$ 4.8	$\leq$ 4.8
No. of Fuel Rod Locations	49	63 or 64	62	60 or 61	59
Fuel Clad O.D. (in.)	$\geq$ 0.5630	$\geq$ 0.4840	$\geq$ 0.4830	$\geq$ 0.4830	$\geq$ 0.4930
Fuel Clad I.D. (in.)	$\leq$ 0.4990	$\leq$ 0.4295	$\leq$ 0.4250	$\leq$ 0.4230	$\leq$ 0.4250
Fuel Pellet Dia. (in.)	$\leq$ 0.4910	$\leq$ 0.4195	$\leq$ 0.4160	$\leq$ 0.4140	$\leq$ 0.4160
Fuel Rod Pitch (in.)	$\leq$ 0.738	$\leq$ 0.642	$\leq$ 0.641	$\leq$ 0.640	$\leq$ 0.640
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Water Rods (Note 10)	0	1 or 0	2	1 - 4 (Note 6)	5
Water Rod Thickness (in.)	N/A	$\geq$ 0.034	> 0.00	> 0.00	$\geq$ 0.034
Channel Thickness (in.)	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 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.% ^{235}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.% ^{235}U) *(Note 14)*	$\leq$ 4.5 (Note 12)	$\leq$ 4.5 (Note 12)	$\leq$ 4.8	$\leq$ 4.8	$\leq$ 4.8
No. of Fuel Rod Locations	76	76	72	92/78 (Note 7)	91/83 (Note 8)
Fuel Clad O.D. (in.)	$\geq$ 0.4170	$\geq$ 0.4430	$\geq$ 0.4240	$\geq$ 0.4040	$\geq$ 0.3957
Fuel Clad I.D. (in.)	$\leq$ 0.3640	$\leq$ 0.3860	$\leq$ 0.3640	$\leq$ 0.3520	$\leq$ 0.3480
Fuel Pellet Dia. (in.)	$\leq$ 0.3530	$\leq$ 0.3745	$\leq$ 0.3565	$\leq$ 0.3455	$\leq$ 0.3420
Fuel Rod Pitch (in.)	$\leq$ 0.572	$\leq$ 0.572	$\leq$ 0.572	$\leq$ 0.510	$\leq$ 0.510
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Water Rods (Note 10)	5	5	1 (Note 5)	2	1 (Note 5)
Water Rod Thickness (in.)	$\geq$ 0.0120	$\geq$ 0.0120	$\geq$ 0.0320	$\geq$ 0.030	> 0.00
Channel Thickness (in.)	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 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.% ^{235}U) *(Note 14)*	$\leq$ 4.8	$\leq$ 4.7 (Note 13)	$\leq$ 4.6 (Note 12)
No. of Fuel Rod Locations	96	92/78 (Note 7)	96/84
Fuel Clad O.D. (in.)	$\geq$ 0.3780	$\geq$ 0.4035	$\geq$ 0.387
Fuel Clad I.D. (in.)	$\leq$ 0.3294	$\leq$ 0.3570	$\leq$ 0.340
Fuel Pellet Dia. (in.)	$\leq$ 0.3224	$\leq$ 0.3500	$\leq$ 0.334
Fuel Rod Pitch (in.)	$\leq$ 0.488	$\leq$ 0.510	$\leq$ 0.512
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Water Rods (Note 10)	5 (Note 9)	2	5 (Note 9)
Water Rod Thickness (in.)	$\geq$ 0.031	$\geq$ 0.030	$\geq$ 0.031
Channel Thickness (in.)	$\leq$ 0.055	$\leq$ 0.120	$\leq$ 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* ~~F~~*f*uel assemblies classified as damaged fuel assemblies, *all assemblies in the MPC* are limited to 4.0 wt.% U-235.

13. *When loading* ~~F~~*f*uel 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.*

Table 2.1.4

DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION

Criterion	**BWR**	**PWR**
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

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

[1] Reference 2.1.7
[2] 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 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment $\leq$ 4.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}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 wt% and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be determined by linear interpolation between the minimum soluble boron concentrations at 4.0 wt% and 5.0 wt% ^{235}U.

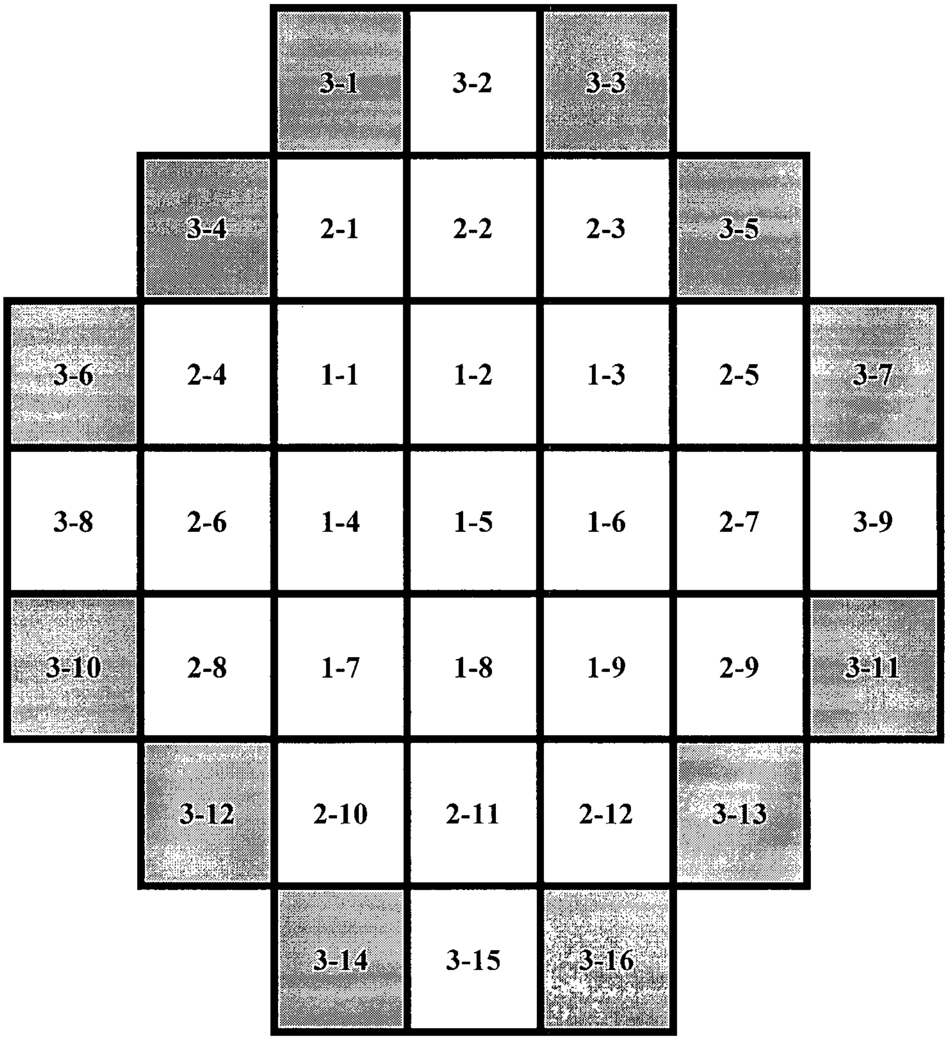


Figure 2.1.1 Location of DFCs for Damaged Fuel or Fuel Debris
in the MPC-37(Shaded Cells)

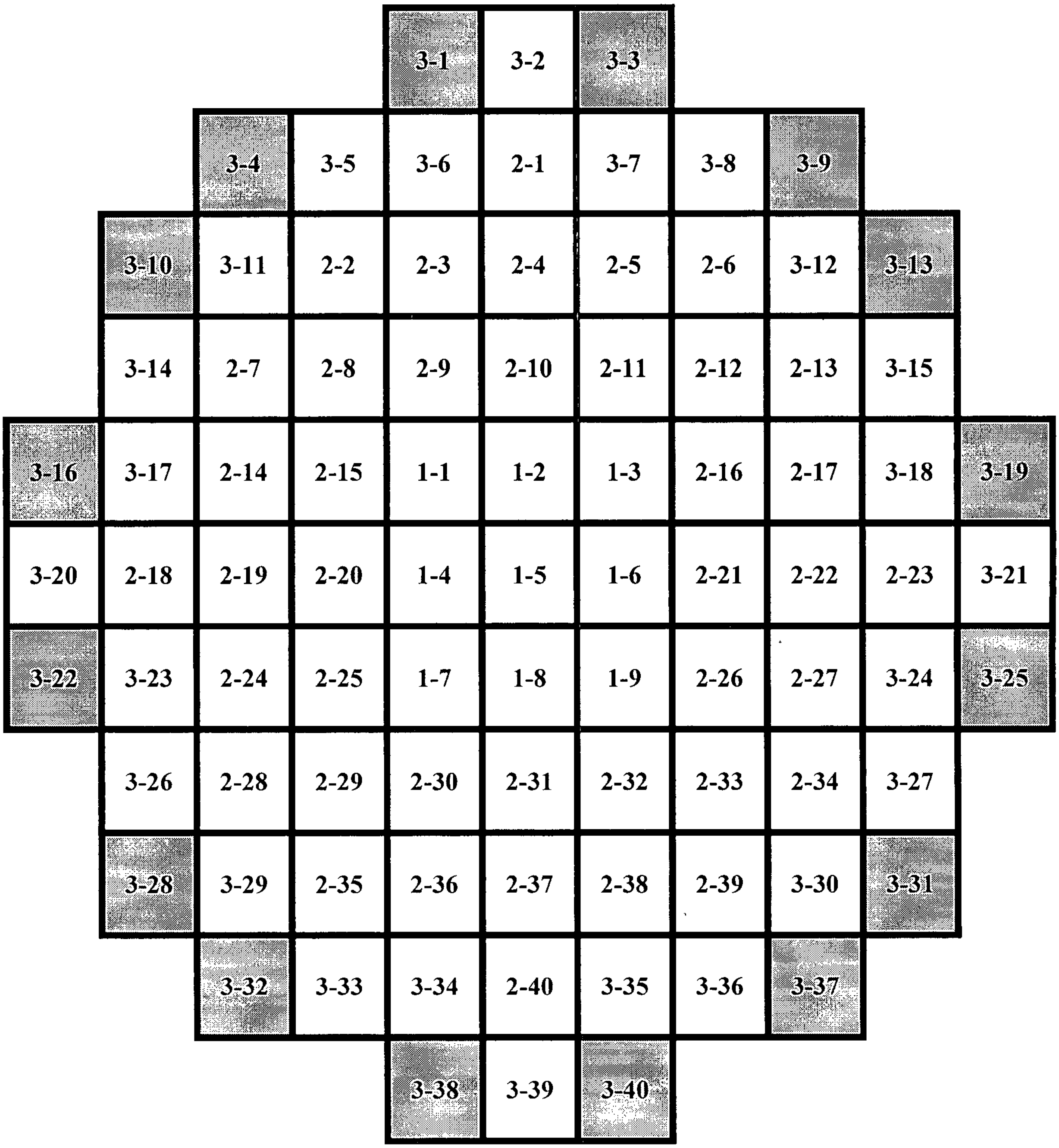


Figure 2.1.2 Location of DFCs for Damaged Fuel or Fuel Debris
in the MPC-89 (Shaded Cells)

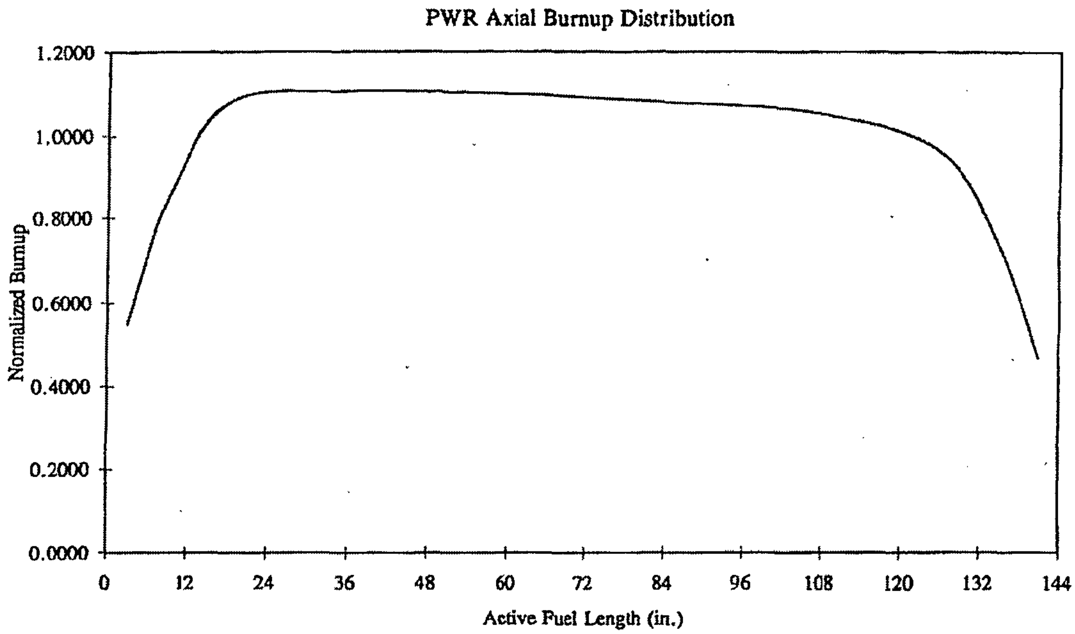


Figure 2.1.3 PWR Axial Burnup Profile with Normalized Distribution

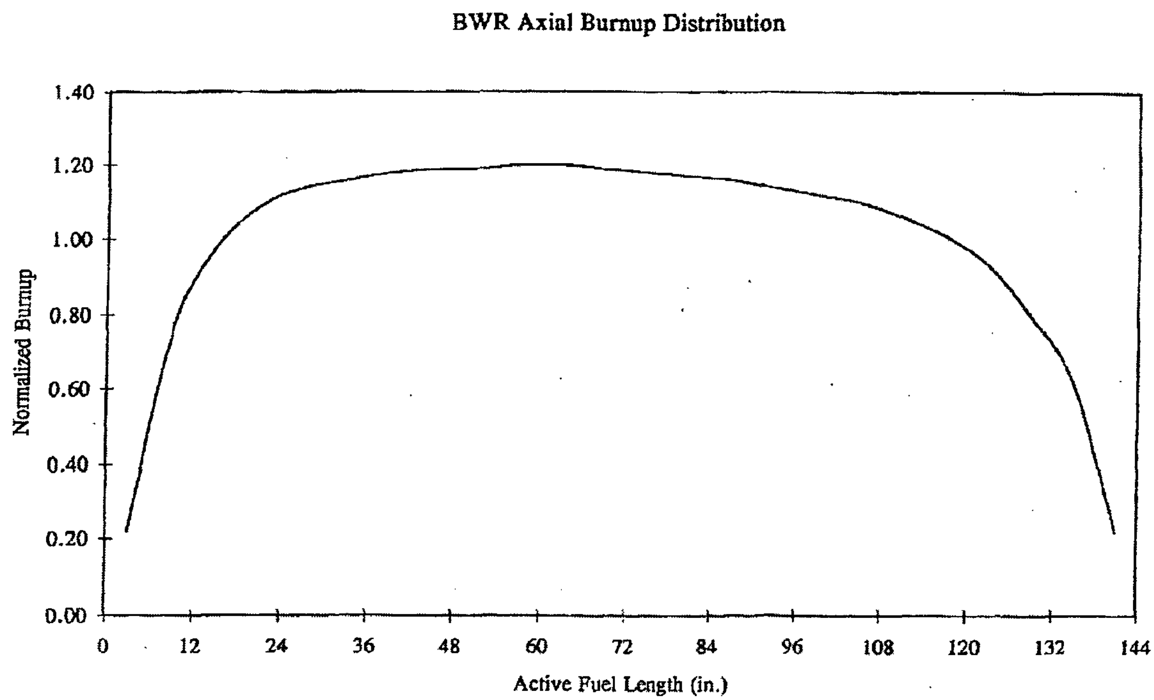


Figure 2.1.4 BWR Axial Burnup Profile with Normalized Distribution

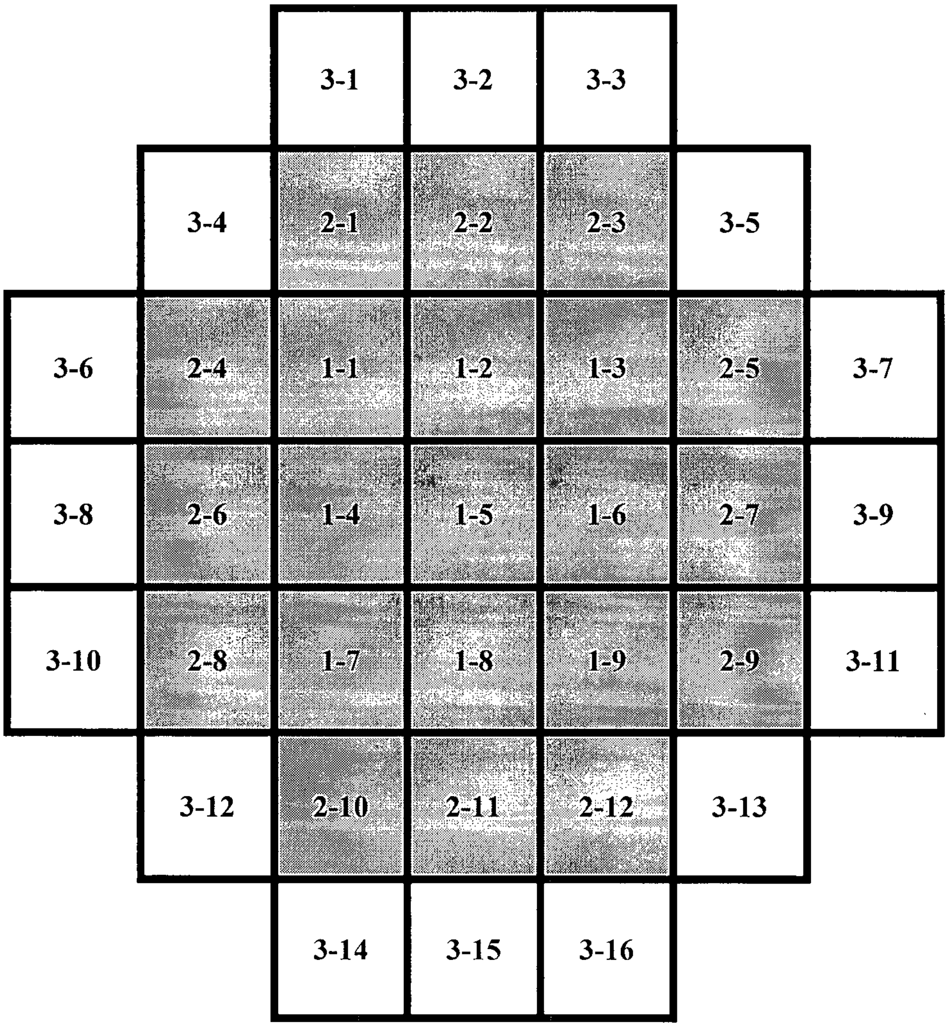


Figure 2.1.5: Location of NSAs, APSRs, RCCAs, CEAs, and CRAs in the MPC-37
(Shaded Cells)

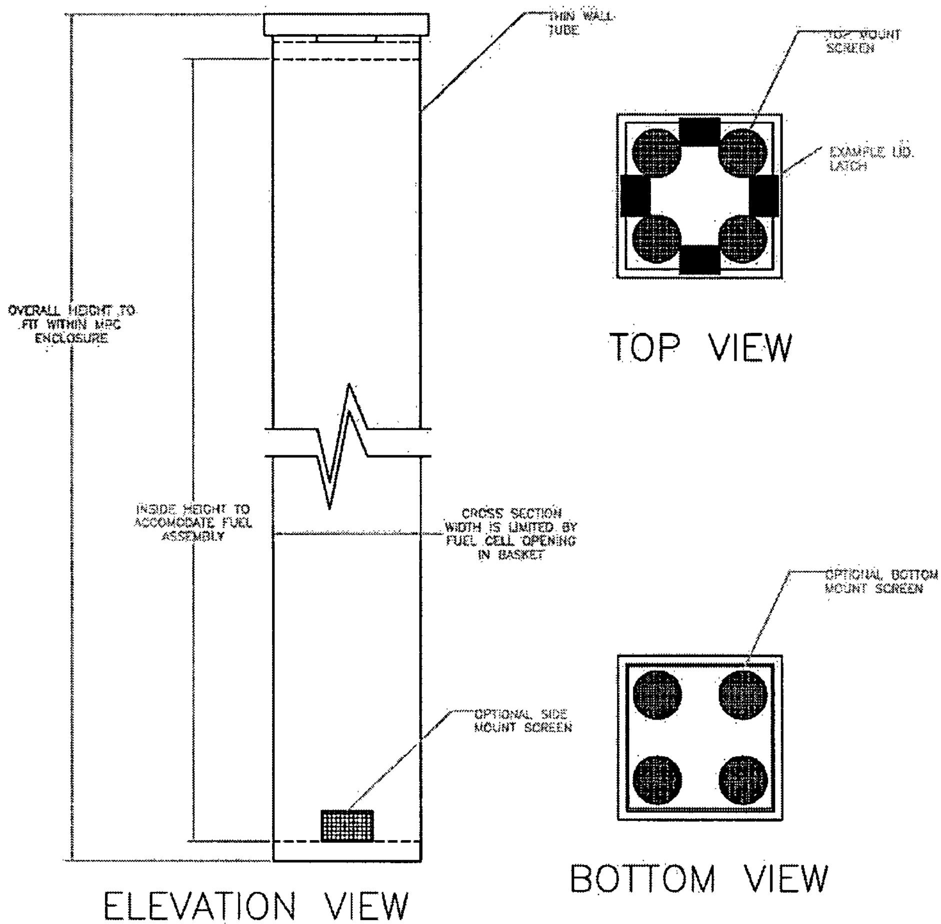


Figure 2.1.6: Damaged Fuel Container (Typical)

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 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) -Thyroid (mrem/yr) -Any Other Critical Organ (mrem/yr)	 25 75 25
DESIGN BASIS ACCIDENT: -TEDE (rem) -DDE + CDE to any individual organ or tissue (other than lens of the eye) (rem) -Lens dose equivalent (rem) -Shallow dose equivalent to skin or any extremity (rem)	 5 50 15 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	~~150~~*300*
Inlet and Outlet Vents	~~250~~*300*
Top of the Lid (Horizontal Surface *at approximate center*)	~~60~~*30*

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)† 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 <u>all</u> 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 compliance with ISG-11 and with NUREG-1536 guidelines, subject to the exceptions and clarifications discussed in Chapter

* 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]).

† Defined as nuclear fuel that is used to produce energy in a commercial nuclear reactor (See Glossary).

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(a), Pattern A and Table 1.2.4) and in the other case maximizes heat dissipation (1.2.3(b), Pattern B).* ~~in each region are guided by the following considerations of~~ *The* ALARA *maximized fuel loading is guided by the following considerations:* ~~and preservation of fuel integrity in long-term storage.~~

- Region *1*~~I~~: Located in the core region of the basket is permitted to store fuel with medium specific heat load.
- Region *2*~~II~~: 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*~~III~~: 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 ~~the peak fuel cladding temperature in the MPC as well as the~~ 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. ~~By limiting the SHL in the core region of the basket,~~ *This has the salutary effect of minimizing* core temperature gradients in the radial direction *and* ~~are minimized, thus minimizing~~ thermal stresses in the fuel and fuel basket.

The salutary consequences of ~~this~~ *all* regionalized loading arrangements become evident from the computed peak cladding temperatures in this chapter, which show ~~a large~~ margin to the ISG-11 limit discussed earlier.

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

Attachment 3 to Holtec Letter 5018019
October 13, 2011

to allow for basket thermal growth during all operational modes.

The most important contributor to minimizing thermal stresses and maximizing heat transmission within the fuel basket is its material of construction (Metamic-HT) which has approximately ten times the thermal conductivity of the stainless steel material used in the stainless steel baskets in the HI-STORM 100 System [4.1.8]. The Metamic-HT plates in the HI-STORM FW MPCs are also considerably thicker than their counterparts in the stainless baskets, resulting in an additional enhancement in conduction heat transfer.

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	7 atm (absolute)
Normal Ambient Temperature	Table 2.2.2
Helium Backfill Pressure	Table 4.4.8

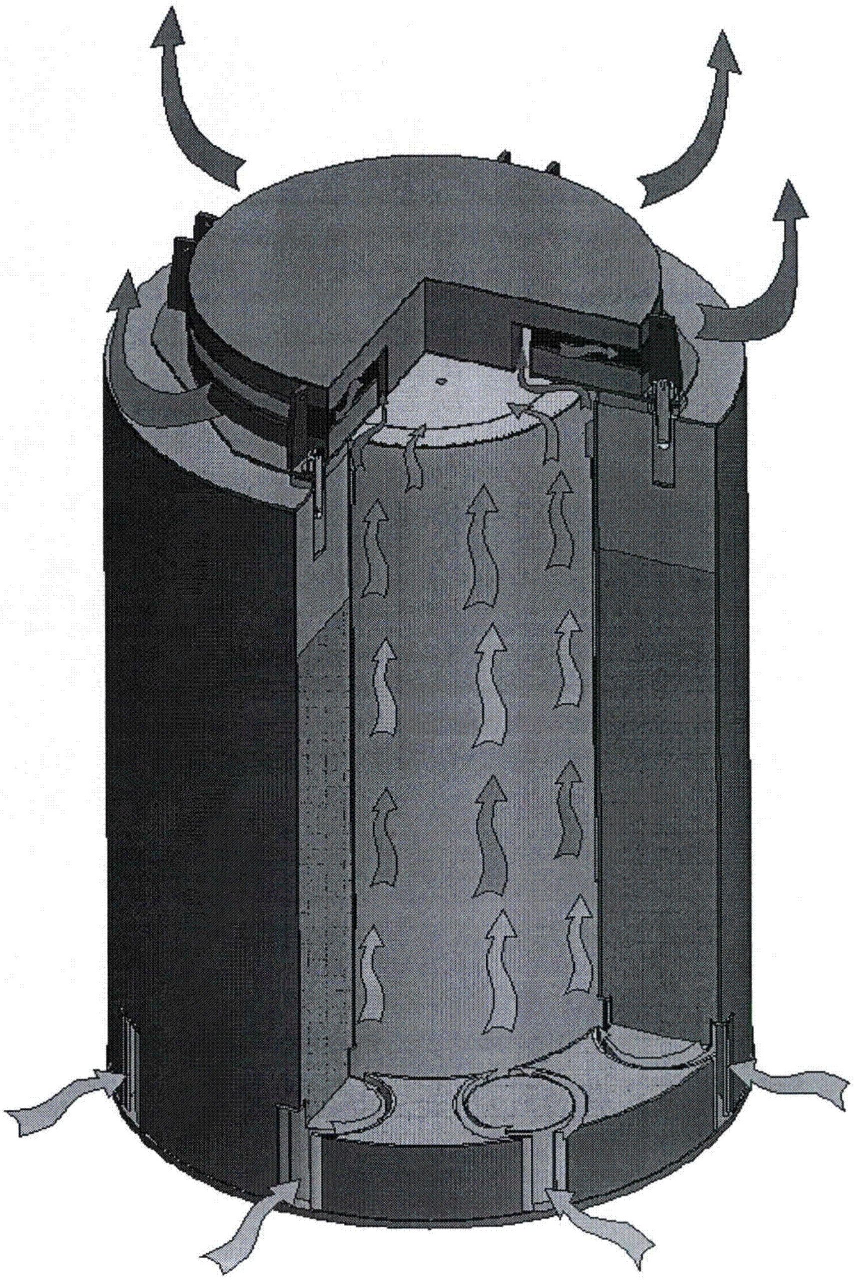


Figure 4.1.1: Ventilation Flow in the HI-STORM FW System

Attachment 3 to Holtec Letter 5018019
October 13, 2011

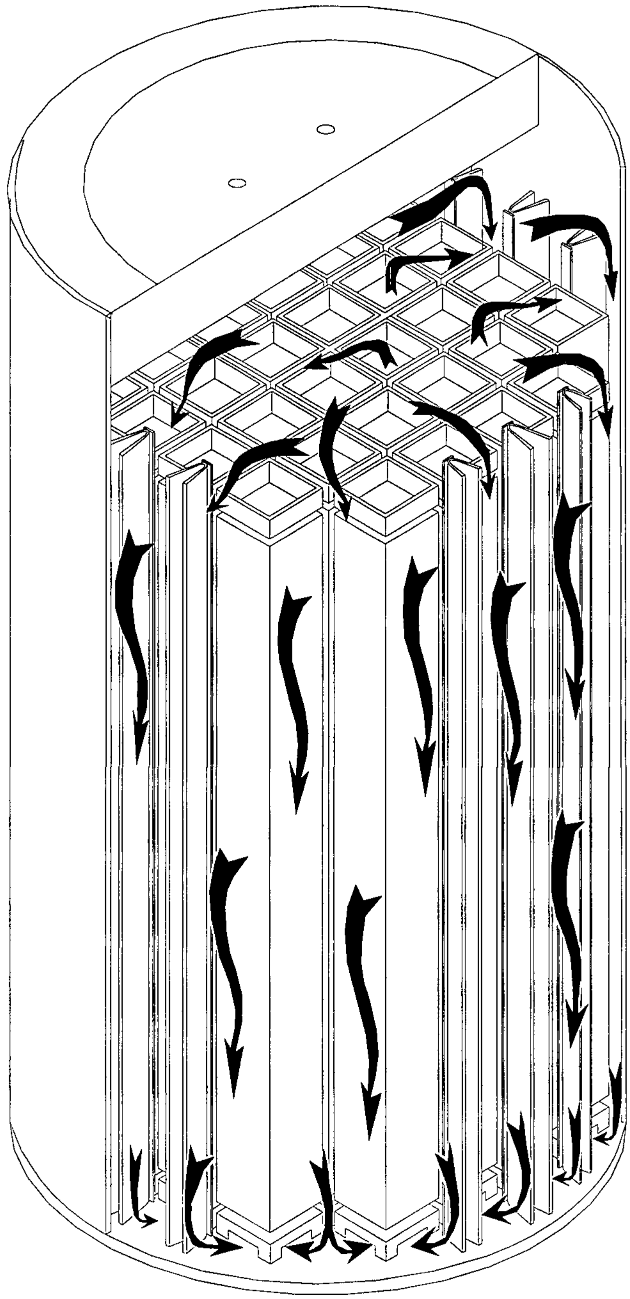


Figure 4.1.2: *Illustration of* MPC Internal Helium Circulation

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*, 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† is applied to painted surfaces. The solar absorbtivity, α_s of paints are generally low. The NASA technical publication [4.2.20] reports α_s in the range of 0.03 to 0.54. For a robustly bounding analysis α_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×Pr is expressed as $L^3\Delta TZ$, where L is height of the overpack, ΔT 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.

* 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.
† 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_2	Note 1	NUREG [4.2.17]	Rust [4.2.4]	Rust [4.2.4]
Stainless Steel (machined forgings)[Note 2]	Kern [4.2.5]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Stainless Steel Plates[Note 3]	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 [Appendix 1.B]	Test Data [4.2.6]	Test Data [4.2.6]	Test Data [4.2.6]
Aluminum *Alloy 2219*	Test Data [Appendix 1.B]	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.B.2			
Aluminum *Alloy 2219***	69.3	69.3	69.3	69.3

* 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.

** Conservatively assumed to be constant for the entire range of temperatures.

Table 4.2.3*

SUMMARY OF FUEL ELEMENT COMPONENTS
THERMAL CONDUCTIVITY DATA

Zircaloy Cladding		**Fuel (UO_2)**	
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

* 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***	See Appendix 1.B, Table 1.B.2
Aluminum *Alloy 2219****	See Appendix 1.B, Table 1.B.2

* See Table 4.2.1 for cited references.

** Lower bound value from the cited references in Table 4.2.1.

*** Metamic-HT and Aluminum shim surfaces are hard anodized to yield high emissivities. Surface emissivity of hard anodized surfaces is reported in Appendix 1.B.

Table 4.2.5 DENSITY AND HEAT CAPACITY PROPERTIES SUMMARY*		
Material	**Density (lbm/ft^3)**	**Heat Capacity (Btu/lbm-°F)**
Helium	(Ideal Gas Law)	1.24
Air	(Ideal Gas Law)	0.24
Zircaloy	409	0.0728
Fuel (UO_2)	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.B.2	Table 1.B.2
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 ($ft^{-3}{}^{\circ}F^{-1}$)***
40	2.1×10^{6}
140	9.0×10^{5}
240	4.6×10^{5}
340	2.6×10^{5}
440	1.5×10^{5}
* 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. The neutron absorber material used in MPC baskets for criticality control (Metamic-HT) is stable in excess of 1000°F*. Accordingly 1000°F is conservatively adopted as the short-term temperature limit for neutron absorber materials. The overpack concrete, the primary function of which is shielding, will maintain its structural, thermal and shielding properties provided that American Concrete Institute (ACI) guidance on temperature limits (see Appendix 1.D in reference [4.1.8]) is followed.

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.

* B_4C is a refractory material that is unaffected by high temperature (on the order of 1000°F) and aluminum is solid at temperatures in excess of 1000°F.

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_4C 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 (a) & (b) (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. The Metamic-HT neutron absorber plates contain 10% (min.) Boron Carbide in an aluminum matrix reinforced with nanoparticles of alumina to provide criticality control, while maximizing heat conduction capabilities (see Appendix 1.B).

Thermal analysis of the HI-STORM FW System is performed for *all*~~limiting~~ 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 lowerbound, upperbound and reference-height MPCs.

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].

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:

<u>Fuel Basket 3D Model</u>
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 three 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: Short Fuel	Minimum Height MPC-37	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:

$$h = 0.19\,(\Delta T)^{1/3}\ (\text{Vertical}, GrPr > 10^9)$$
$$h = 0.18\,(\Delta T)^{1/3}\ (\text{Horizontal Cylinder}, GrPr > 10^9)$$
$$(\text{in conventional U.S. units})$$

Laminar range:

$$h = 0.29\,\left(\frac{\Delta T}{L}\right)^{1/4}\ (\text{Vertical}, GrPr < 10^9)$$
$$h = 0.27\,\left(\frac{\Delta T}{D}\right)^{1/4}\ (\text{Horizontal Cylinder}, GrPr < 10^9)$$
$$(\text{in conventional U.S. Units})$$

where ΔT 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 2.6×10^5 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 ΔT (~10°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 absorbtivity. To model insolation heating a reasonably bounding absorbtivity 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.

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-ω 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 a heat shield to protect the inner shell and concrete from radiation heating by the emplaced MPC. The heat shield, inner and outer shells and concrete are explicitly modeled.

i) To maximize lateral resistance to heat dissipation in the fuel basket, 0.4 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 absorbtivity 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-ω 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 μm) with a bounding low conductivity (conductivity of helium). The crud resistance increases the clad temperature by a very small amount (~0.1°F) [4.1.8]. Accordingly this effect is neglected in the thermal evaluations.

4.4.1.2 Fuel Assembly 3-Zone Flow Resistance Model*

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 (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 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:

$$\Delta P/L = D\mu V \qquad \text{(Eq. 1)}$$

where $\Delta P/L$ is the hydraulic pressure loss per unit length, D is the flow resistance coefficient, μ 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† is replaced by porous media. As discussed below the porous media length is partitioned in three zones with discrete flow resistances.

* 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].
† 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.

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.

4.4.1.3 Bounding Flow Resistance Data

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). 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

Attachment 3 to Holtec Letter 5018019
October 13, 2011

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.

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*~~design maximum~~ heat load *patterns* defined in Tables 1.2.3* 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.

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* *Pattern A defined in Table 1.2.3(a) is the limiting fuel storage pattern (See Subsection 4.4.4.1).*

4.4.1.6 Grid Sensitivity Studies

The discretization of the MPC and the HI-STORM/MPC annulus region must be sufficiently dense to insure a converged solution. Because the flow field in the annulus is in the transition and turbulent regimes, the grid size and layout are critical to insuring a converged solution. In the MPC internal space, however, the flow is uniformly laminar (no laminar boundary layer to turbulent zone transition effects) and therefore, the grid size is relatively unimportant. The sensitivity study was accordingly performed on the annulus region outside the MPC and the grid size in the axial direction within the MPC. All sensitivity analyses were carried out for the case of *limiting*~~47.05 kW~~ *Pattern A* ~~design maximum~~ heat load *in*~~for~~ the~~(limiting)~~ MPC-37 canister *(See Subsection 4.4.4.1).*

a. The HI-STORM FW annulus grid sensitivity results are tabulated below.

Run No	Number of Radial Cells	y^+	PCT (°C)	Permissible Limit (°C)	Clad Temperature Margin (°C)
1	6	21	353	400	47
2	10	5	357	400	43
3	11	4	364	400	36
4	12	3	376	400	24
5	17	0.7	375	400	25

Note 1: The y^+ reported in the third column above is a measure of grid adequacy provided by the FLUENT code. Values of $y^+ \sim 1$ indicate an adequate level of mesh refinement is reached to resolve the viscosity affected region near the wall.
Note 2: The annulus grid is refined in two ways, namely, by increasing the number of radial cells and also by clustering the cells near the MPC and overpack innershell walls.

As can be seen from the above table, the thermal solution is quite sensitive to the grid density in the annulus region. The above results show that Run No 5 is reasonably converged. To provide further assurance of convergence, the sensitivity results are evaluated in accordance with the ASME Journal procedure for control of numerical accuracy [4.4.3]. Towards this end the Grid Convergence Index (GCI), which is a measure of the solution uncertainty, is computed. The GCI for the finest grid (i.e. 17 radial cells) computes to be 1.3×10^{-5}% which provides further assurance of grid convergence. Having obtained grid convergence in the annulus region, the Run No 5 grid is adopted for further grid sensitivity studies below.

b. The results of axial grid refinement in the fueled region are summarized below.

Grid Refinement	Number of Axial Cells	PCT (°C)	Permissible Limit (°C)	Clad Temperature Margin (°C)
Baseline run	84[Note 1]	375	400	25

(Run No 5 adopted from above)				
Refined Grid	101	376	400	24
Note 1: As explained below the baseline grid is adopted for thermal evaluation of the HI-STORM FW.				

The above results show that the solution is essentially unchanged by further grid refinement in the axial direction. This result is in keeping with the fact that the flow field in the MPC internal space is uniformly laminar. Based on the above results, Run No 5 grid layout is adopted for the thermal analysis of the HI-STORM FW.

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^2

Attachment 3 to Holtec Letter 5018019
October 13, 2011

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^2

The above numerical exercise shows that $A_o >> A_{min}$ 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 co-axially in it) is constructed for the15 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	707	45
Hypothetical Square Cavity Thermal Model	705*	47
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 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. 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 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.

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(a), 1.2.3(b) and 1.2.4.*

- *The limiting fuel temperatures are reached under the Pattern A thermal loading condition defined in Table 1.2.3(a) 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 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.

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.

4.4.5 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 atmospheres (absolute) 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*s* *are*~~is~~ defined *below:*~~wherein each fuel storage location is assumed to be generating heat at 80% of the Design Basis fuel assembly heats defined in Tables~~

Attachment 3 to Holtec Letter 5018019
October 13, 2011

~~1.2.3 and 1.2.4 and the MPC sufficiently backfilled to yield 6 atmospheres absolute pressure.~~

(i) *MPC-37 under 80% Pattern A Heat Load (Table 1.2.3(a))*
(ii) *MPC-37 under 90% Pattern A Heat Load (Table 1.2.3(a))*
(iii) *MPC-89 under 80% Design Heat Load (Table 1.2.4)*
(iv) *MPC-37 under vacuum drying threshold heat load in Table 4.5.1**.
(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*s (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.

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

* *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.*

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.

~~(i)~~*(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 (shortest MPC height and ~~design~~ 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).

Attachment 3 to Holtec Letter 5018019
October 13, 2011

Table 4.4.1

EFFECTIVE FUEL PROPERTIES UNDER BOUNDING FUEL STORAGE CONFIGURATIONS[Note 1]

	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
Thermal Inertia Properties				
	Density (lb/ft^3)		Heat Capacity (Btu/lb-°F)[Note 2]	
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_2 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).

Attachment 3 to Holtec Letter 5018019
October 13, 2011

Table 4.4.2 RESULTS OF SCREENING CALCULATIONS UNDER NORMAL STORAGE CONDITIONS	
Storage Scenario	**Peak Cladding Temperature, °C (°F)**
MPC-37 Minimum Height* Reference Height Maximum Height	 **353 (667)** 342 (648) 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 baseline mesh.

* Bounding scenario adopted in this Chapter for all thermal evaluations.

Table 4.4.3 MAXIMUM TEMPERATURES IN ~~THE LIMITING~~ HI-STORM FW ~~STORAGE SCENARIO~~ UNDER LONG-TERM NORMAL STORAGE*	
Component	**Temperature, °C (°F)** ***Pattern A / Pattern B***
Fuel Cladding	375 (707) / *371 (700)*
MPC Basket	361 (682) / *356(673)*
Basket Periphery	297 (567) / *295(563)*
Aluminum Basket Shims	276 (529) / *272(522)*
MPC Shell	246 (475) / *245(473)*
MPC Lid[Note 1]	243 (469) / *238(460)*
Overpack Inner Shell	128 (262) / *129(264)*
Overpack Outer Shell	60 (140) / *63(145)*
Overpack Body Concrete[Note 1]	88 (190) / *90(194)*
Overpack Lid Concrete[Note 1]	113 (235) / *113(235)*
Area Averaged Air outlet†	104 (219) / *105(221)*
Note 1: Maximum section average temperature is reported.	

* The temperatures reported in this table (all for shortest fuel scenario of MPC-37) are below the design temperatures specified in Table 2.2.3, Chapter 2. *These temperatures bound MPC-89 temperatures.*

† 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^3)**	**MPC-89 (ft^3)**
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~~5.5~~
Normal: intact rods 1% rods rupture	 98.0/*98.5* 99.1/*99.7*	 98.2~~0~~ 98.7~~6~~
Off-Normal (10% rods rupture)	109.1/*109.6*	103.*8*~~9~~
Accident (100% rods rupture)	195.5/*196.0*	154.*7*~~6.3~~
* 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. ** Conservatively assumed at the Tech. Spec. maximum value (see Table 4.4.8). ~~*** Conservatively the MPC-37 cavity average temperature is adopted for pressure calculations.~~		

Attachment 3 to Holtec Letter 5018019
October 13, 2011

Table 4.4.6

SUMMARY OF HI-STORM FW DIFFERENTIAL THERMAL EXPANSIONS

Gap Description	Cold Gap U (in)	Differential Expansion δ_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 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*	40.3	46.6
MPC-37 Pattern B	*40.5*	*46.8*
MPC-89~~***~~	*42.0~~40.3~~*	*48.5~~46.6~~*

* 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.

~~*** Conservatively the MPC-37 cavity average temperature is adopted for pressure calculations.~~

Attachment 3 to Holtec Letter 5018019
October 13, 2011

Table 4.4.8 MPC HELIUM BACKFILL PRESSURE SPECIFICATIONS		
MPC	**Item**	**Specification**
MPC-37 *Pattern A*	Minimum Pressure	42.*0*~~5~~ 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*~~5.5~~ psig @ 70°F Reference Temperature
~~* Conservatively the MPC-37 cavity average temperature is adopted for pressure calculations.~~		

Table 4.4.9 MAXIMUM HI-STORM FW TEMPERATURES AT ELEVATED SITES*	
Component	**Temperature, °C (°F)**
Fuel Cladding	374 (705)
MPC Basket	360 (680)
Aluminum Basket Shims	275 (527)
MPC Shell	246 (475)
MPC Lid[Note 1]	242 (468)
Overpack Inner Shell	126 (259)
Overpack Body Concrete[Note 1]	86 (187)
Overpack Lid Concrete[Note 1]	112 (234)
Note 1: Maximum section average temperature is reported.	

* 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[Note 1]

MPC	Item	Specification
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	
MPC-37 *(80% of Pattern A in Table 1.2.3(a))* Region 1 Cells Region 2 Cells Region 3 Cells Total	0.904 kW/assy 1.424 kW/assy 0.776 kW/assy 37.64 kW
MPC-37 (90% of Pattern A in Table 1.2.3(a)) *Region 1 Cells* *Region 2 Cells* *Region 3 Cells* *Total*	*1.017 kW/assy* *1.602 kW/assy* *0.873 kW/assy* *42.34 kW*
MPC-89 *(80% of Table 1.2.4)* 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.	

Attachment 3 to Holtec Letter 5018019
October 13, 2011

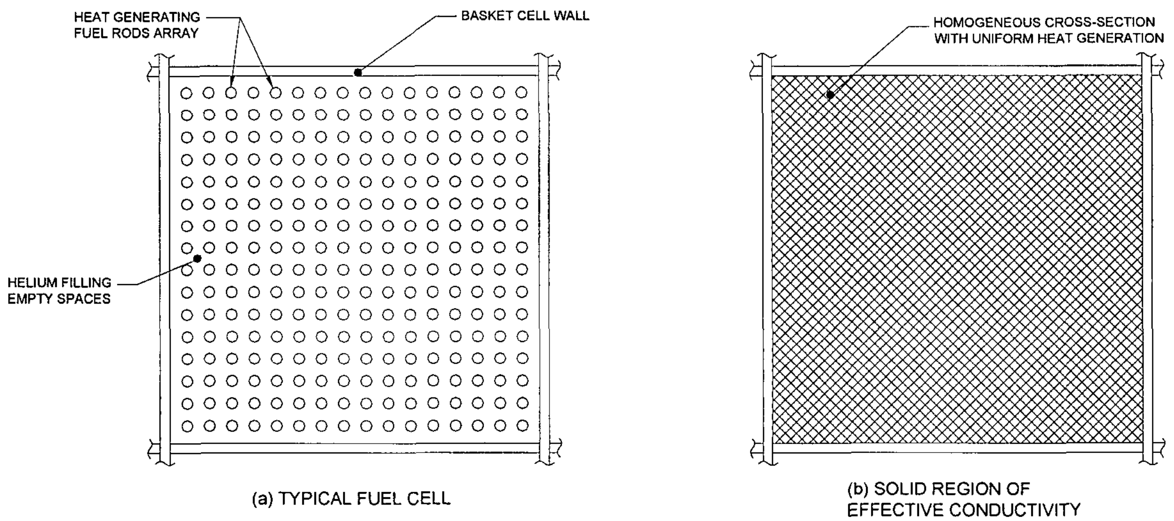


Figure 4.4.1: Homogenization of the Storage Cell Cross-Section

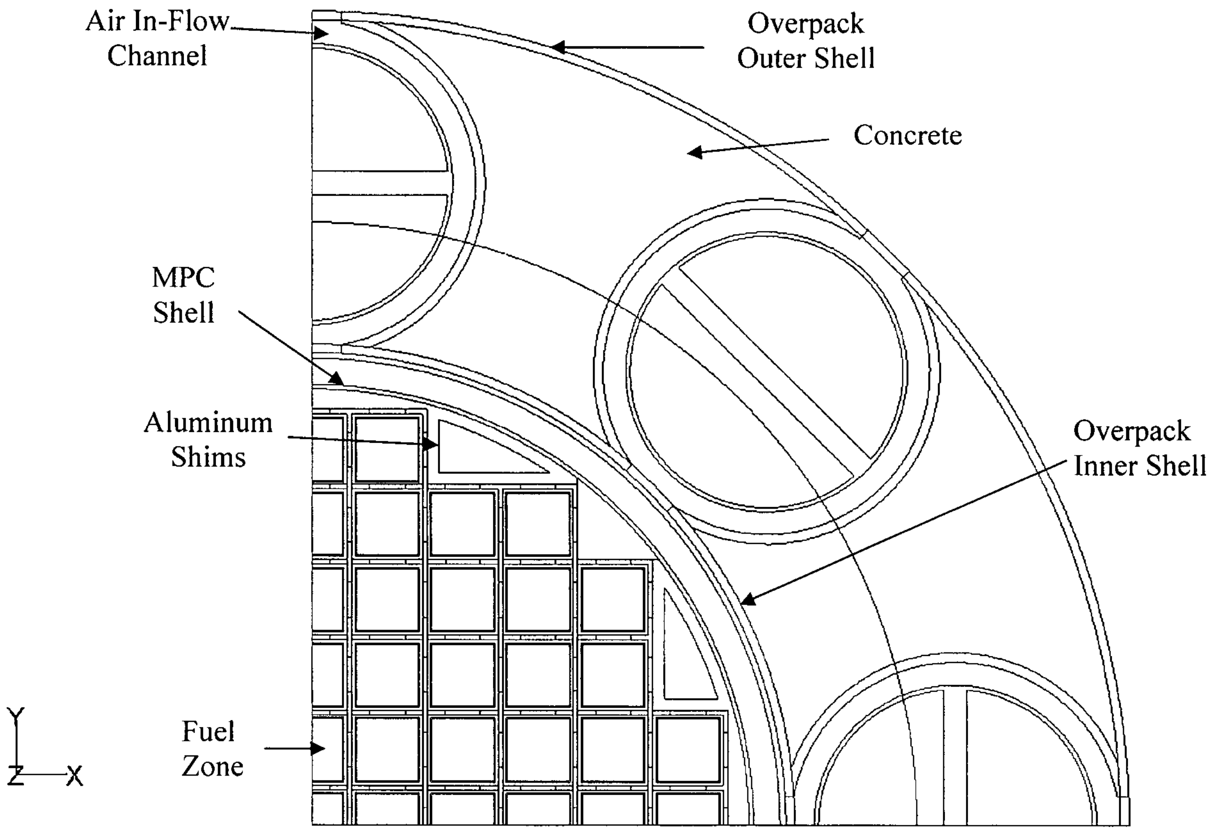


Figure 4.4.2: Planar View of HI-STORM FW MPC-89 Quarter Symmetric 3-D Model

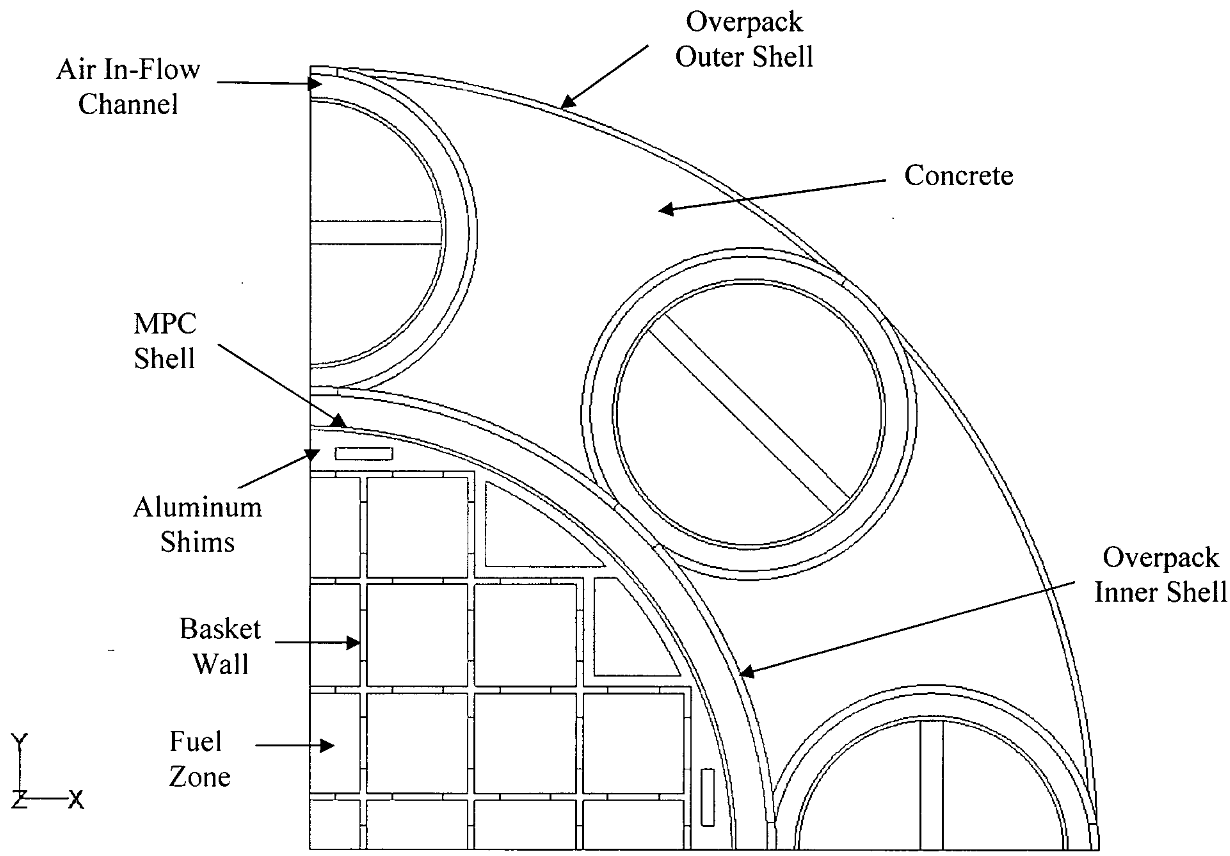


Figure 4.4.3: Planar View of HI-STORM FW MPC-37 Quarter Symmetric 3-D Model

Attachment 3 to Holtec Letter 5018019
October 13, 2011

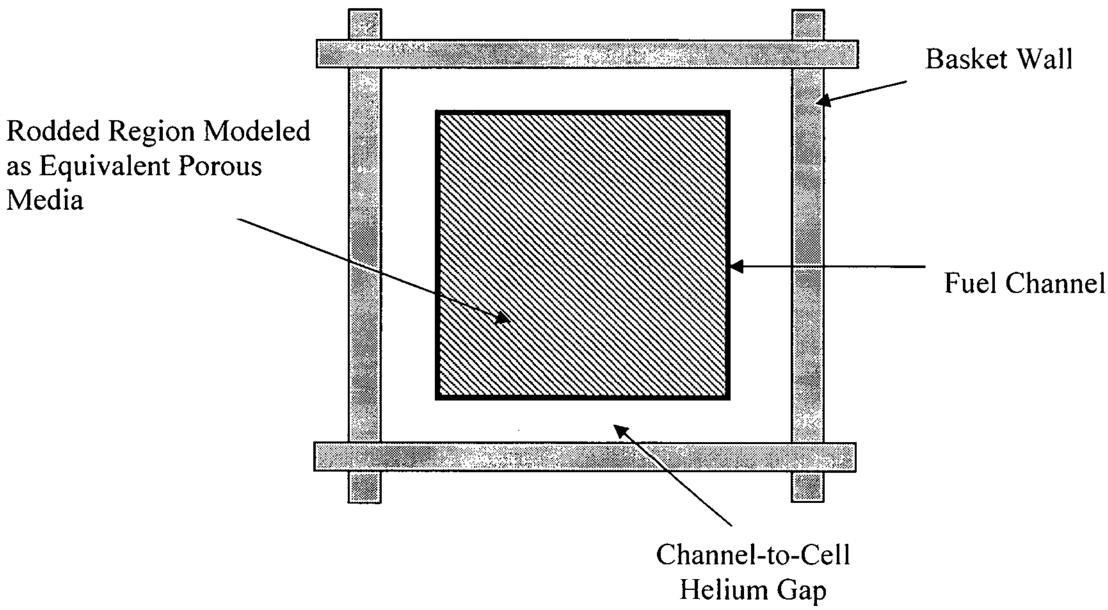
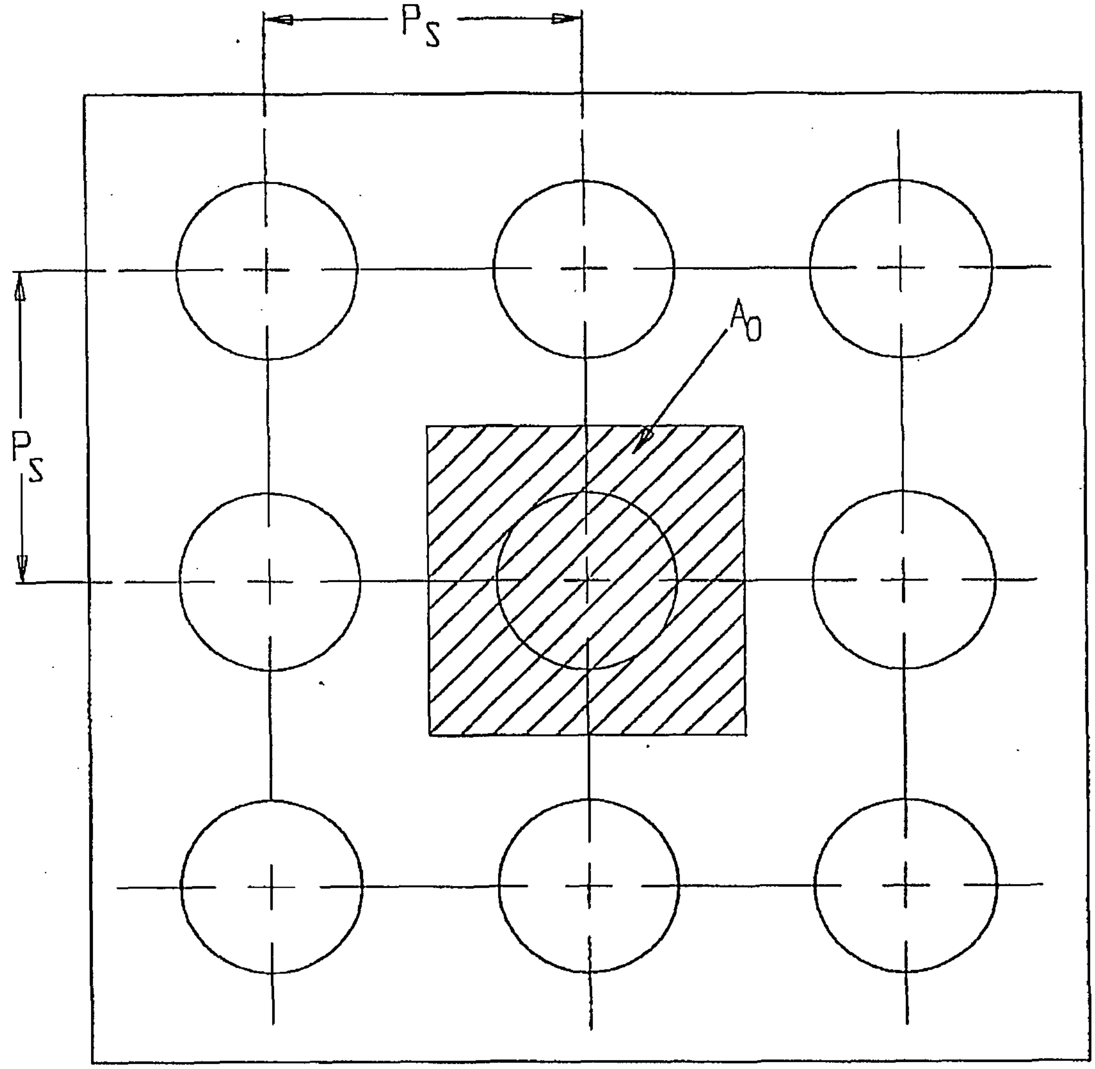


Figure 4.4.4: Closeup View of the MPC-89 Channeled Fuel Spaces



Legend:
P_s: Cask pitch
A_o: Tributary area

Figure 4.4.5: Illustration of a Centrally Located Cask in a Cask Array

Attachment 3 to Holtec Letter 5018019
October 13, 2011

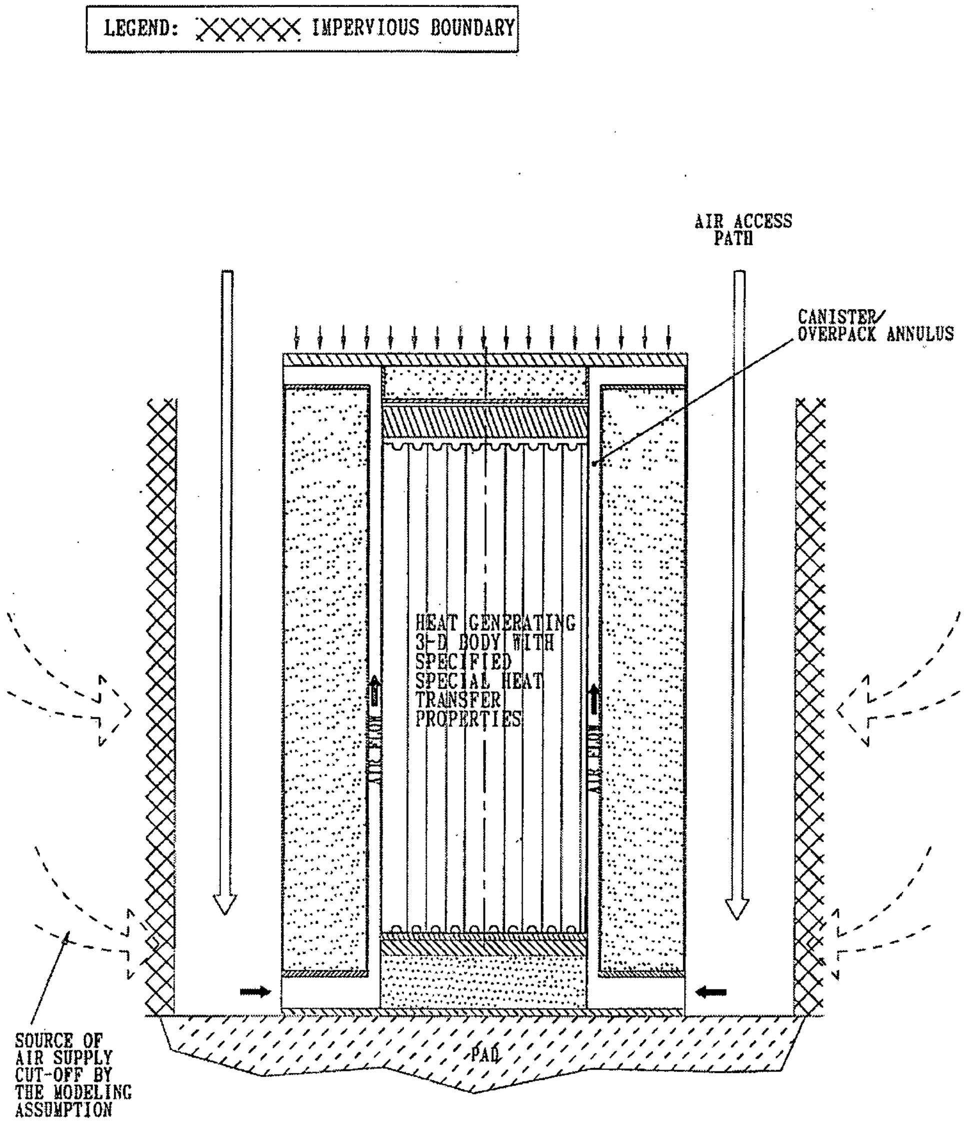


Figure 4.4.6: Illustration of the Hypothetical Square Cavity Thermal Model

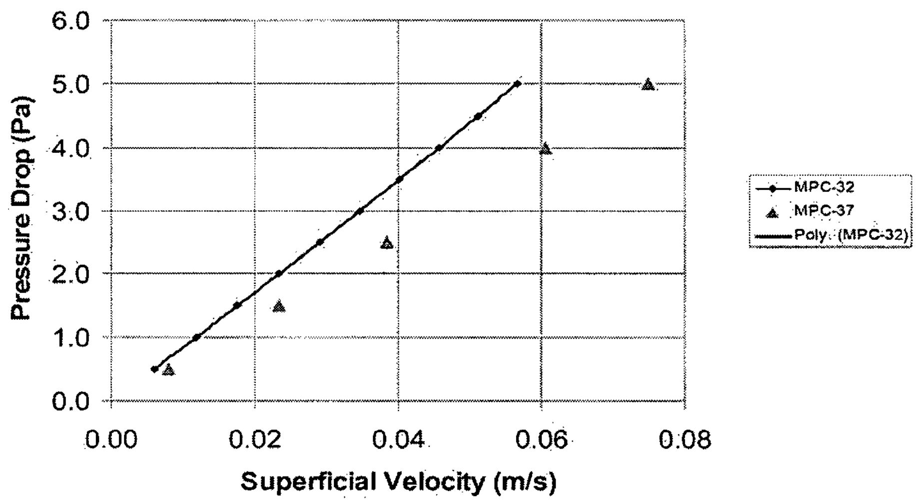


Figure 4.4.7: Storage Cell Pressure Drop as a Function of In-Cell Helium Velocity

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 decease 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 absorbtivity 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 0.4 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.

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*~~at design maximum~~ 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 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.

Attachment 3 to Holtec Letter 5018019
October 13, 2011

A 3-D FLUENT thermal model of the MPC is constructed in the same manner as described in Section 4.4. 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*~~design~~ heat load (*Pattern A in* Tables 1.2.3*(a)* 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

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.

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. The time limits are conservatively computed under an assumed adiabatic temperature rise of the cask with design heat load and understated thermal inertia of the cask defined in Table 4.5.3. The computed time limits are tabulated in Table 4.5.4.

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

Attachment 3 to Holtec Letter 5018019
October 13, 2011

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*~~design basis~~ 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 demoisturizer, a compressor, and a pre-heater is utilized to extract moisture from the MPC cavity through repeated displacement of its contained helium, accompanied by vigorous flow turbulation. Demoisturization to the3 torr vapor pressure criteria required by NUREG 1536 is assured by verifying that the helium temperature exiting the demoisturizer 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

Attachment 3 to Holtec Letter 5018019
October 13, 2011

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.

4.5.5 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

Storage Zone [Note 1]	**MPC-37 (kW)**	**MPC-89 (kW)**
Region 1	0.8	0.35
Region 2	0.97	0.35
Region 3	0.97	0.44
MPC Heat Load	34.36	34.75
Note 1: Storage zones are defined in Figures 1.2.1 and 1.2.2.		

Table 4.5.2

HI-TRAC VW TRANSFER CASK STEADY STATE MAXIMUM TEMPERATURES[Note 1]

Component	Temperature, °C (°F)
Fuel Cladding	388 (730)
MPC Basket	375 (707)
Basket Periphery	305 (581)
Aluminum Basket Shims	283 (541)
MPC Shell	254 (489)
MPC Lid[Note 2]	244 (471)
HI-TRAC VW Inner Shell	144 (291)
HI-TRAC VW Radial Lead Gamma Shield	143 (289)
Water Jacket Bulk Water	134 (273)

Note 1: The temperatures tabulated herein are updated to reflect the changes in helium backfill pressures in Table 4.4.8.
Note 2: Maximum section average temperature is reported.

Attachment 3 to Holtec Letter 5018019
October 13, 2011

Table 4.5.3 HI-TRAC VW TRANSFER CASK LOWERBOUND WEIGHTS AND THERMAL INERTIAS			
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

Table 4.5.4 MAXIMUM ALLOWABLE TIME FOR WET TRANSFER OPERATIONS	
Initial temperature °F	**Time Duration (hr)**
100	14.2
110	12.9
120	11.6
130	10.4
140	9.1
150	7.8

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	101.9

Table 4.5.6 MAXIMUM TEMPERATURES OF MPC-37 DURING VACUUM DRYING CONDITIONS		
Component	**Temperatures @DB Heat Load Note 1** **°C (°F)**	**Temperatures @ Threshold Heat Load Note 2** **°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 Note 3	107 (225)	100 (212)
Note 1: Addresses vacuum drying of Moderate Burnup Fuel under *limiting*~~Design Basis~~ 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.		

Attachment 3 to Holtec Letter 5018019
October 13, 2011

Table 4.5.7 MAXIMUM TEMPERATURES OF MPC-89 DURING VACUUM DRYING CONDITIONS		
Component	**Temperatures @DB Heat Load[Note 1] °C (°F)**	**Temperatures @ Threshold Heat Load[Note 2] °C (°F)**
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[Note 3]	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[Note 1] 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).

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.

<u>(a) HI-STORM FW Fire</u>

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^2×°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^2 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 (shortest 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^2-hr)
h_{fc} = Forced Convection Heat Transfer Coefficient (4.5 Btu/ft^2-hr-°F)
σ = Stefan-Boltzmann Constant
T_A = Fire Temperature (1475°F)

C= Conversion Constant (460 (^{o}F to ^{o}R))
T_S = Surface Temperature (^{o}F)
ε = 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^2 (Btu/ft^2-hr))
h_S = Natural Convection Heat Transfer Coefficient (Btu/ft^2-hr-^{o}F)
T_S = Surface Temperature (^{o}F))
T_A = Ambient Temperature (^{o}F)
σ = Stefan-Boltzmann Constant
ε = Surface Emissivity
C= Conversion Constant (460 (^{o}F to ^{o}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^2 and has a depth of 0.696 in. From this depth and the fuel consumption rate of 0.15 in/min, the fire duration τ_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 τ_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-^{o}F)
ΔT = Permissible temperature rise (^{o}F)
Q = Decay heat load (Btu/hr)

Substituting the parameters in Table 4.6.6, the minimum available burial time is computed as 85.1 hours. The co-incident 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 upto 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 ^{o}C (^{o}F)**	**Partial Inlets Duct Blockage ^{o}C (^{o}F)**
Fuel Cladding	386 (727)	385 (725)
MPC Basket	372 (702)	371 (700)
Aluminum Basket Shims	287 (549)	285 (545)
MPC Shell	257 (495)	257 (495)
MPC Lid*	254 (489)	252 (486)
Overpack Inner Shell	139 (282)	141 (286)
Overpack Outer Shell	71 (160)	62 (144)
Overpack Body Concrete*	99 (210)	95 (203)
Overpack Lid Concrete*	124 (255)	122 (252)

Table 4.6.2

HI-STORM FW FIRE AND POST-FIRE ACCIDENT ANALYSIS RESULTS

Component	Initial Condition °C (°F)	End of Fire Condition °C (°F)	Post-Fire Cooldown °C (°F)
Fuel Cladding	375 (707)	375 (707)	377 (711)
MPC Basket	361 (682)	361 (682)	363 (685)
Basket Periphery	297 (567)	297 (567)	299 (570)
Aluminum Basket Shims	276 (529)	276 (529)	278 (532)
MPC Shell	246 (475)	251 (484)	251 (484)
MPC Lid Note 1	243 (469)	245 (473)	245 (473)
Overpack Inner Shell	128 (262)	140 (284)	140 (284)
Overpack Outer Shell	60 (140)	340 (644) Note 3	340 (644) Note 3
Overpack Body Concrete Note 1	88 (190)	100 (212)	100 (212)
Overpack Lid Concrete Note 1	113 (235)	125 (257)	125 (257)

Note 1: Maximum section average temperature is reported.
Note 2: The temperatures tabulated herein are obtained by adding a conservatively postulated temperature increment to a baseline mesh thermal solution that suitably bounds the effects of grid sensitivity evaluated in Para 4.4.1.6 of Subsection 4.4.1.
Note 3: Surface average temperature is reported.

Table 4.6.3 HI-TRAC VW JACKET WATER LOSS MAXIMUM TEMPERATURES	
Component	**Temperature** **^oC (^oF)**
Fuel Cladding	432 (810)
MPC Basket	416 (781)
Basket Periphery	342 (648)
Aluminum Basket Shims	314 (597)
MPC Shell	290 (554)
MPC Lid*	263 (505)
HI-TRAC VW Inner Shell	205 (401)
HI-TRAC VW Radial Lead Gamma Shield	204 (399)

* Maximum section average temperature is reported.

Table 4.6.4 EXTREME ENVIRONMENTAL CONDITION MAXIMUM HI-STORM FW TEMPERATURES	
Component	**Temperature*** **°C (°F)**
Fuel Cladding	400 (752)
MPC Basket	386 (727)
Basket Periphery	322 (612)
Aluminum Basket Shims	301 (574)
MPC Shell	271 (520)
MPC Lid[Note 1]	268 (514)
Overpack Inner Shell	153 (307)
Overpack Outer Shell	85 (185)
Overpack Body Concrete[Note 1]	113 (235)
Overpack Lid Concrete[Note 1]	138 (280)
Average Air Outlet	129 (264)
Note 1: Maximum section average temperature is reported.	

* 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.

Attachment 3 to Holtec Letter 5018019
October 13, 2011

Table 4.6.5 RESULTS OF HI-STORM FW 32-HOURS BLOCKED INLET DUCTS THERMAL ANALYSIS		
Component*	**Initial Condition °C (°F)**	**Final Condition °C (°F)**
Fuel Cladding	375 (707)	484 (903)
MPC Basket	361 (682)	468 (874)
Basket Periphery	297 (567)	404 (759)
Aluminum Basket Shims	276 (529)	380 (716)
MPC Shell	246 (475)	358 (676)
MPC Lid[Note 1]	243 (469)	313 (595)
Overpack Inner Shell	128 (262)	247 (477)
Overpack Outer Shell	60 (140)	105 (221)
Overpack Body Concrete[Note 1]	88 (190)	130 (266)
Overpack Lid Concrete[Note 1]	113 (235)	165 (329)
Note 1: Maximum section average temperature is reported. Note 2: The temperatures tabulated herein are obtained by adding a conservatively postulated temperature increment to a baseline mesh thermal solution that suitably bounds the effects of grid sensitivity evaluated in Para 4.4.1.6 of Subsection 4.4.1.		

* For a bounding evaluation, temperatures are computed at the lowerbound helium backfill pressure defined in Table 4.4.8. Temperatures of limiting components reported.

Table 4.6.6 SUMMARY OF INPUTS FOR BURIAL UNDER DEBRIS ANALYSIS	
Thermal Inertia Inputs*: M (Lowerbound HI-STORM FW Weight) Cp (Carbon steel heat capacity)†	 215000 kg 419 J/kg-°C
Clad initial temperature[Note 1]	390°C
Q (Decay heat)	47.05 kW
ΔT (clad temperature margin)‡	160°C
Note 1: Initial temperature conservatively postulated to bound the maximum cladding temperature.	

* Thermal inertia of fuel is conservatively neglected.

† 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).

‡ 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*	111.6
Partial Blockage of Inlet Ducts	99.9
Accident Conditions	
HI-TRAC VW fire accident	103.3
Extreme Ambient Temperature	103.1
100% Blockage of Air Inlets	116.4
Burial Under Debris	130.8
HI-TRAC VW Jacket Water Loss	109.5

* 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[†]: 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), 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 Table*s* 1.2.3 *and 1.2.4*), the ~~packaging~~ *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 ~~regionalized~~ loading configuration, in the MPC-37~~,~~ ~~*as described in Section 1.2*~~. ~~*As for*~~ ~~and t~~*The* MPC-89 ~~, there is only~~*has one heat load pattern with a three-region* ~~*regionalized*~~ *loading configuration* ~~as described in Section 2.1.*1.2*~~. The~~is~~ regionalized storage pattern*s* ~~is~~ *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]:

[†] 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.

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. α,n reactions in fuel materials
    3. Secondary neutrons produced by fission from subcritical multiplication
    4. γ,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 extend, 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 ~~regionalized~~ 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 ~~regionalized~~ -loading configuration - Loading Pattern A and Loading Pattern B. An important difference between Pattern A and Pattern B 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 ~~would be~~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 ~~w~~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 ~~regionalized~~ 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.*

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.

~~As stated in Section 5.0, the HI-STORM FW System offers a three-region regionalized loading configuration. Based on this configuration, 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 Tables 1.2.3 and 1.2.4). A more detailed description of the benefits of regionalized loading can be found in Section 5.4. 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. The burnup and cooling time combination is selected as a representative average for the entire basket. This way, dose rates on the outer radial surface of the cask and at distances from the cask will be slightly overestimated, since the basket periphery has assemblies with below average burnup and/or above average cooling times. For the top of the cask, the average burnup and cooling time is expected to result in more realistic dose rate. The representative burnup and cooling time is then selected based on the other limiting fuel parameter, namely the average heat load per assembly.~~

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 increased dose values, due to the non-linear relationship between burnup and neutron source term. On the other hand, for very short cooling times, with corresponding lower burnups, dose locations that are gamma-dominated may show increased 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, representative burnup and cooling time values are selected for all calculations for normal conditions. The selected values are shown in Table 5.0.1. 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	
MPC-37	**MPC-89**
45,000 MWD/MTU *4.5* ~~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	
MPC-37	**MPC-89**
65,000 MWD/MTU ~~10~~ *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 5.1.2 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, adjacent, and one meter 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.

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.

Table 5.1.1

DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS
MPC-37 DESIGN BASIS FUEL
45,000 MWD/MTU AND *4*.5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE HI-TRAC VW						
1	*975*~~839~~	*25*~~24~~	*808*~~757~~	*67*~~66~~	*1874*~~1686~~	*1874*~~1686~~
2	*2939*~~2537~~	*75*~~74~~	*<1*~~<1~~	*154*~~151~~	*3169*~~2763~~	*3169*~~2763~~
3	*20*~~16~~	*5*~~5~~	*339*~~318~~	*6*~~6~~	*371*~~345~~	*561*~~535~~
4	*98*~~81~~	*1*~~1~~	*530*~~496~~	*225*~~220~~	*854*~~798~~	*1147*~~1091~~
5	*940*~~806~~	*3*~~2~~	*2074*~~1942~~	*1022*~~1003~~	*4038*~~3753~~	*4038*~~3753~~
ONE METER FROM THE HI-TRAC VW						
1	*695*~~604~~	*12*~~11~~	*99*~~92~~	*30*~~29~~	*835*~~736~~	*835*~~736~~
2	*1382*~~1195~~	*22*~~22~~	*10*~~9~~	*58*~~56~~	*1472*~~1282~~	*1474*~~1285~~
3	*268*~~234~~	*6*~~6~~	*142*~~133~~	*9*~~9~~	*425*~~382~~	*501*~~458~~
4	*80*~~68~~	*<1*~~<1~~	*295*~~276~~	*73*~~72~~	*449*~~417~~	*613*~~581~~
5	*470*~~403~~	*1*~~1~~	*1129*~~1057~~	*297*~~291~~	*1897*~~1752~~	*1897*~~1752~~

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}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.1.2

DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS
MPC-89 DESIGN BASIS FUEL
45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW					
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
ONE METER FROM THE HI-TRAC VW					
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}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
HI-STORM FW Overpack					
45,000 MWD/MTU AND 4.5-YEAR COOLING					
Annual Dose (mrem/year)	*18*~~15~~	*15*~~13~~	*23*~~19~~	*11*~~9~~	*14*~~12~~
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.

Attachment 3 to Holtec Letter 5018019
October 13, 2011

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,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ONE METER FROM HI-TRAC VW						
65,000 MWD/MTU AND ~~10~~*8*-YEAR COOLING						
2 (Accident Condition)	*1735* ~~1735~~ ~~1404~~	*3*~~3~~	*13*~~10~~	*2651*~~2458~~	*4403*~~3875~~	*4407*~~3880~~
2 (Normal Condition)	*893*~~716~~	*50*~~46~~	*7*~~5~~	*122*~~113~~	*1071*~~880~~	*1074*~~882~~
100 METERS FROM HI-TRAC VW						
65,000 MWD/MTU AND ~~10~~*8*-YEAR COOLING						
2 (Accident Condition)	*0.7*~~0.6~~	*<0.1*~~0.1~~	*0.1*~~0.1~~	*1.4*~~1.3~~	*2.3*~~2.1~~	*2.4*~~2.2~~

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}Co from the spacer grids, and ^{60}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,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	*273*~~234~~	*2*~~2~~	*14*~~13~~	*44*	*292*~~253~~	*292*~~253~~
2	*137*~~115~~	*1*~~1~~	<*1*~~<1~~	*1*~~1~~	*141*~~118~~	*141*~~118~~
3 (surface)	*11*~~9~~	*1*~~<1~~	*25*~~24~~	*2*~~2~~	*39*~~36~~	*53*~~50~~
3 (overpack edge)	*13*~~11~~	<*1*~~<1~~	*63*~~59~~	*1*~~<1~~	*78*~~72~~	*113*~~107~~
4 (center)	<*1*~~0.1~~	*1*~~0.5~~	<*1*~~0.4~~	<*1*~~0.1~~	<*4*~~1.1~~	<*4*~~1.4~~
4 (mid)	*1*~~1~~	*1*~~1~~	*44*	*1*~~1~~	*7*~~7~~	*10*~~9~~
4 (outer)	*10*~~8~~	<*1*~~<1~~	*30*~~28~~	<*1*~~<1~~	*42*~~38~~	*59*~~55~~

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}Co from the spacer grids, and ^{60}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

Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
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}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

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	*57*~~48~~	*1*~~<1~~	*44*	*1*~~<1~~	*62*~~54~~	*62*~~54~~
2	*75*~~64~~	*1*~~<1~~	*1*~~<1~~	*1*~~<1~~	*77*~~67~~	*78*~~67~~
3	*6*~~5~~	*<1*~~<1~~	54	*<1*~~<1~~	*13*~~11~~	*15*~~14~~
4 (center)	*0.6*~~0.4~~	*0.3*~~0.3~~	*1.0*~~1~~	*0.2*~~0.2~~	*2.1*~~1.9~~	*2.7*~~2.5~~

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}Co from the spacer grids, and ^{60}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

Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
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}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 ~~10~~*8*-YEAR COOLING

Dose Point Location	**Dose Rate (rem/hr)**	**Accident Duration (days)**	**Total Dose (rem)**	**Regulatory Limit (rem)**	**Time to Reach Regulatory Limit (days)**
2 (Accident Condition)	*2.3E-3*~~2.2E-3~~	*30*~~30~~	*1.66* ~~1.6~~	*5* ~~5~~	*90* ~~94~~

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.

Attachment 3 to Holtec Letter 5018019
October 13, 2011

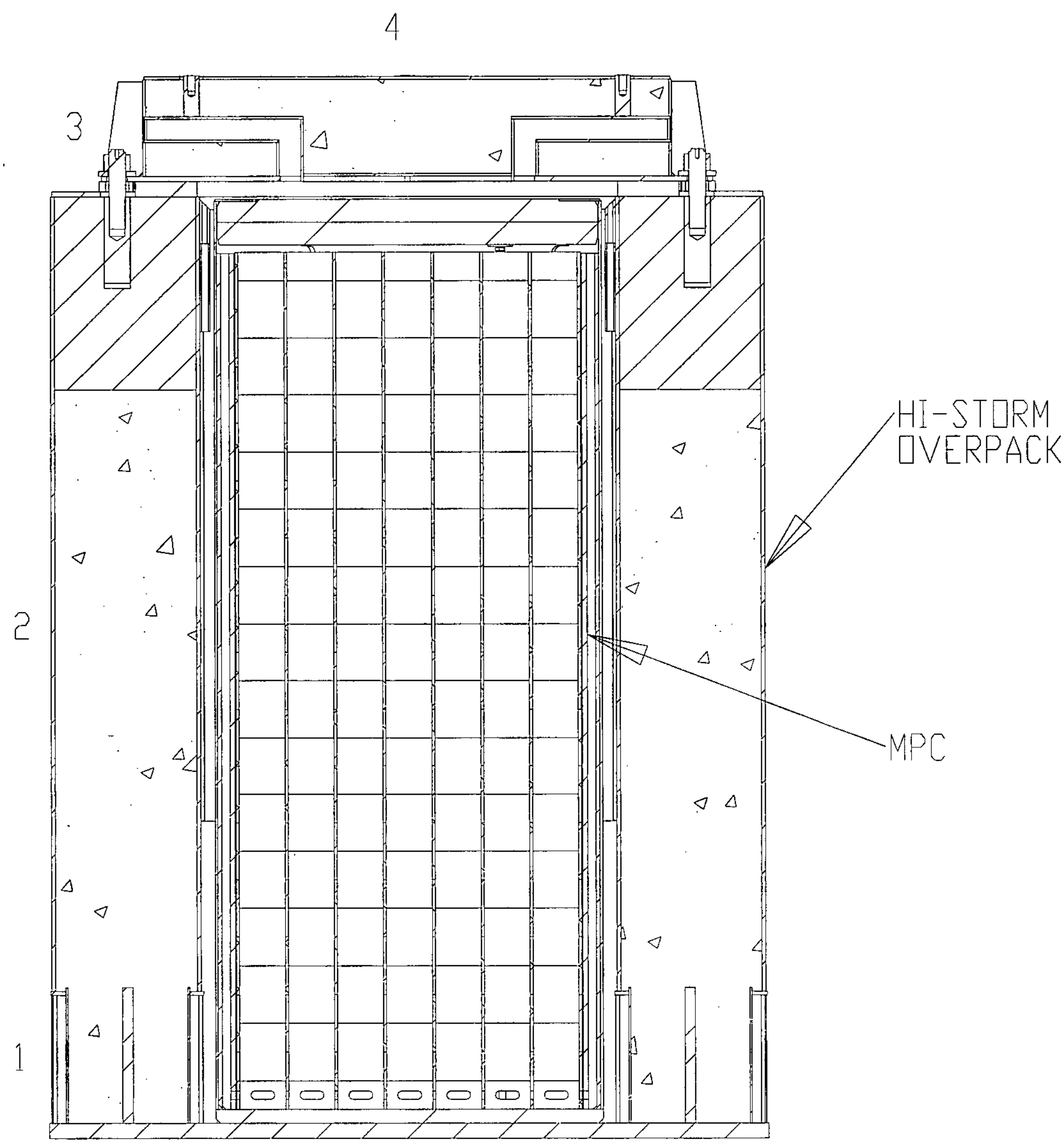


Figure 5.1.1

CROSS SECTION ELEVATION VIEW OF HI-STORM FW OVERPACK WITH DOSE POINT LOCATIONS

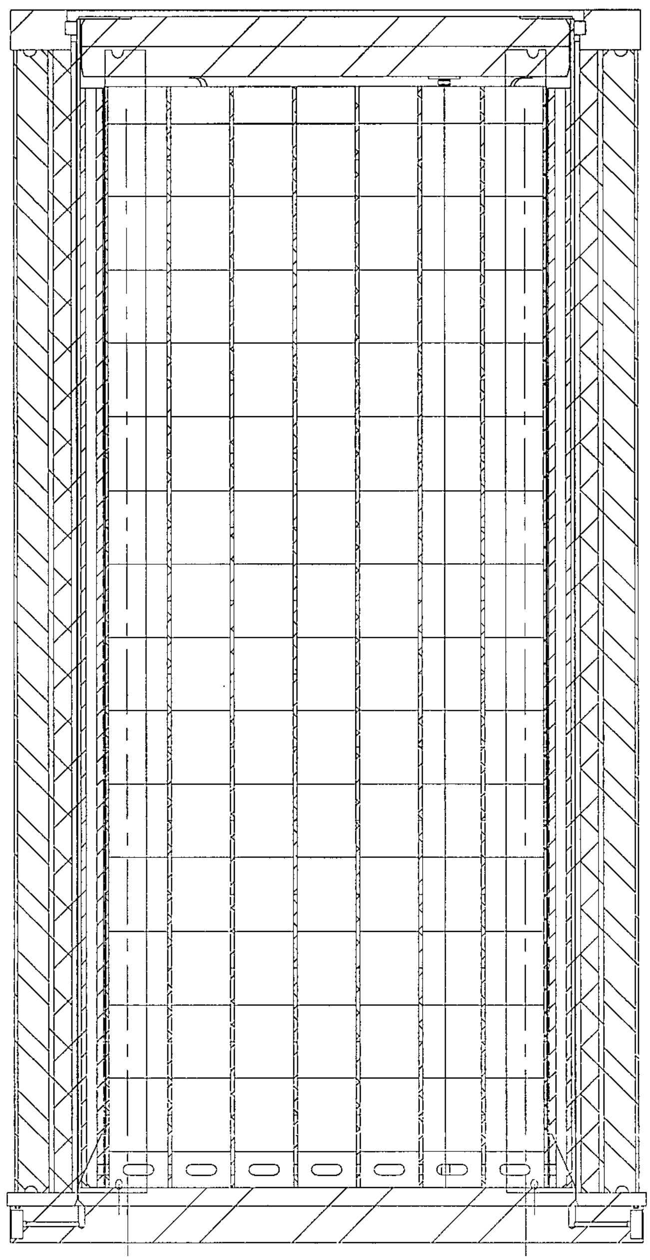
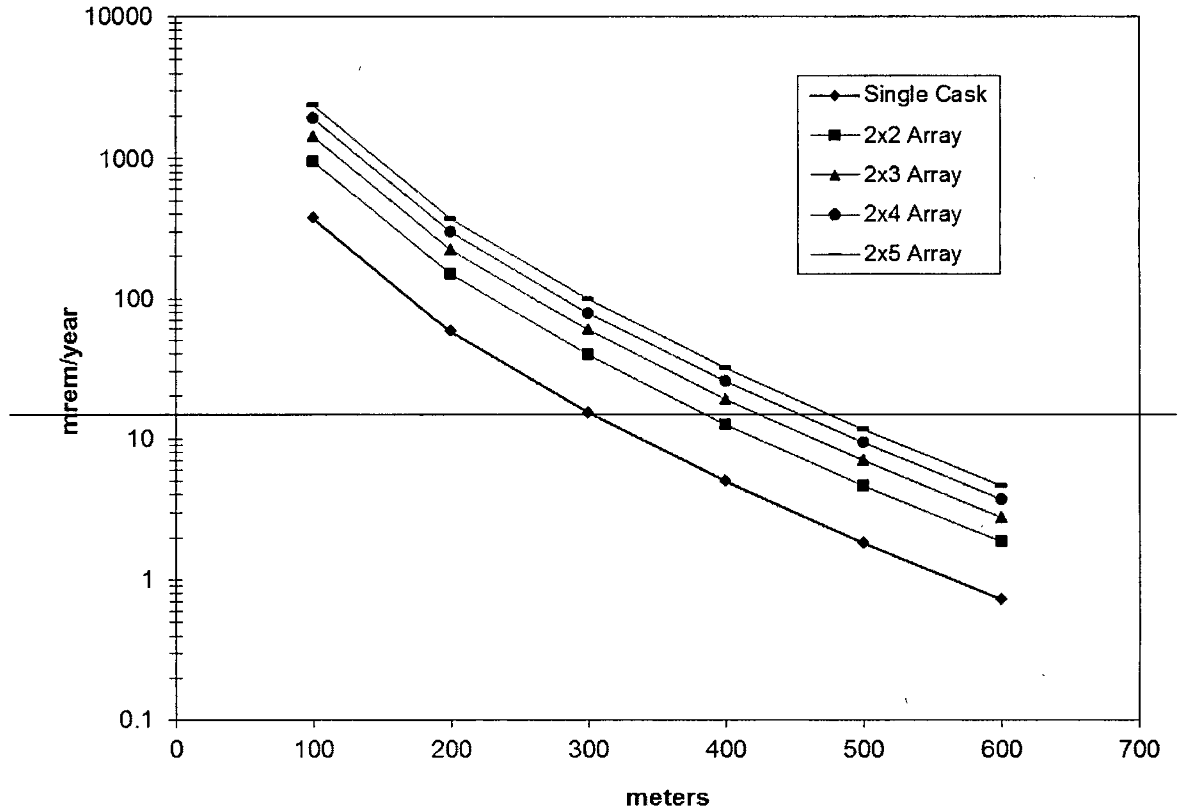


Figure 5.1.2
CROSS SECTION ELEVATION VIEW OF HI-TRAC VW TRANSFER CASK WITH DOSE POINT LOCATIONS



10000
1000
100
10
1
0.1
mrem/year
0
100
200
300
400
500
600
700
meters
Single Cask
2x2 Array
2x3 Array
2x4 Array
2x5 Array

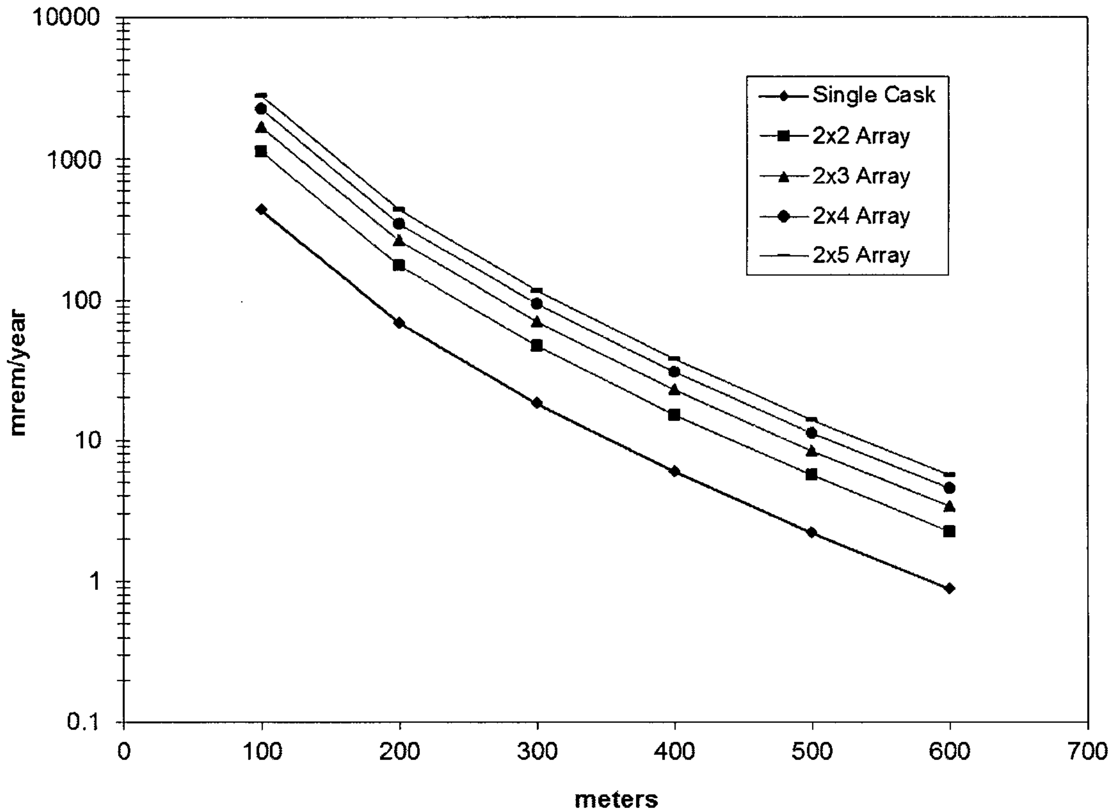


Figure 5.1.3

ANNUAL DOSE VERSUS DISTANCE FOR VARIOUS CONFIGURATIONS OF THE MPC-37 FOR 45,000 MWD/MTU AND *4.5*~~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}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,γ) 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}Co to ^{60}Co. The primary source of ^{59}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}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}Co impurity level in the fuel assembly material. The grams of impurity were then used in ORIGEN-S to calculate a ^{60}Co activity level for the desired burnup and decay time. The methodology used to determine the activation level was developed from Reference [5.2.3] and is described here.

1. The activity of the ^{60}Co is calculated using ORIGEN-S. The flux used in the calculation was the in-core fuel region flux at full power.

2. The activity calculated in Step 1 for the region of interest was modified by the appropriate scaling factors listed in Table 5.2.6. These scaling factors were taken from Reference [5.2.3].

Tables 5.2.7 through 5.2.10 provide the ^{60}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,γ) reactions in the material of the MPC and the overpack. This source of photons is properly accounted for in MCNP when a neutron calculation is performed in a coupled neutron-gamma mode.

5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel, which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in order to obtain conservative source terms, low initial fuel enrichments 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.

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_4C 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, poloniumberyllium, 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	2	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_2	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.522 (96% of theoretical)
Enrichment (w/o ^{235}U)	3.6	3.2
Specific power (MW/MTU)	43.48	30
Weight of UO_2 (kg)††	532.150	213.531
Weight of U (kg)††	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

†† Derived from parameters in this table.

Table 5.2.1 (continued)

DESCRIPTION OF DESIGN BASIS FUEL

	PWR	BWR
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 ~~5~~*4.5*-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	*2.11E+15*~~1.95E+15~~	*3.68E+15*~~3.40E+15~~
0.7	1.0	*7.67E+14*~~6.52E+14~~	*9.02E+14*~~7.67E+14~~
1.0	1.5	*1.74E+14*~~1.52E+14~~	*1.39E+14*~~1.22E+14~~
1.5	2.0	*1.45E+13*~~1.19E+13~~	*8.30E+12*~~6.79E+12~~
2.0	2.5	*1.01E+13*~~6.64E+12~~	*4.47E+12*~~2.95E+12~~
2.5	3.0	*4.05E+11*~~2.88E+11~~	*1.47E+11*~~1.05E+11~~
Total		*3.08E+15*~~2.78E+15~~	*4.73E+15*~~4.30E+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*~~10~~-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	*2.05E+15*~~1.79E+15~~	*3.56E+15*~~3.12E+15~~
0.7	1.0	*4.16E+14*~~2.38E+14~~	*4.89E+14*~~2.80E+14~~

1.0	1.5	*1.30E+14*~~9.79E+13~~	*1.04E+14*~~7.83E+13~~
1.5	2.0	*8.66E+12*~~7.00E+12~~	*4.95E+12*~~4.00E+12~~
2.0	2.5	*6.46E+11*~~1.46E+11~~	*2.87E+11*~~6.50E+10~~
2.5	3.0	*4.49E+10*~~1.24E+10~~	*1.63E+10*~~4.50E+09~~
Total		*2.60E+15*~~2.13E+15~~	*4.16E+15*~~3.48E+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}Co SOURCE

Region	PWR	BWR
Handle	N/A	0.05
Upper End Fitting	0.1	0.1
Gas Plenum Spacer	0.1	N/A
Expansion Springs	N/A	0.1
Gas Plenum Springs	0.2	0.2
Incore Grid Spacer	1.0	1.0
Lower End Fitting	0.2	0.15

Table 5.2.7

CALCULATED MPC-37 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS

Location	**45,000 MWD/MTU and *4*.5-Year Cooling (curies)**
Lower End Fitting	*86.02*~~80.53~~
Gas Plenum Springs	*16.77*~~15.70~~
Gas Plenum Spacer	*11.91*~~11.15~~
Expansion Springs	*NA*~~N/A~~
Incore Grid Spacers	*357.19*~~334.42~~
Upper End Fitting	*57.22*~~53.57~~
Handle	*NA*~~N/A~~

Table 5.2.8

CALCULATED MPC-37 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS

Location	**65,000 MWD/MTU and ~~10~~*8*-Year Cooling (curies)**
Lower End Fitting	*64.89*~~49.87~~
Gas Plenum Springs	*12.65*~~9.72~~
Gas Plenum Spacer	*8.99*~~6.91~~
Expansion Springs	*NA*~~N/A~~
Incore Grid Spacers	*269.46*~~207.09~~
Upper End Fitting	*43.17*~~33.18~~
Handle	*NA*~~N/A~~

Table 5.2.9

CALCULATED MPC-89 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS

Location	45,000 MWD/MTU and 5-Year Cooling (curies)
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}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS

Location	65,000 MWD/MTU and 10-Year Cooling (curies)
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 ~~5~~ *4.5* YEAR COOLING

Lower Energy (MeV)	**Upper Energy (MeV)**	**45,000 MWD/MTU ~~5~~*4.5*-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	*3.05E+07*~~2.99E+07~~
4.0e-01	9.0e-01	*6.64E+07*~~6.52E+07~~
9.0e-01	1.4	*6.63E+07*~~6.51E+07~~
1.4	1.85	*5.30E+07*~~5.20E+07~~
1.85	3.0	*9.88E+07*~~9.69E+07~~
3.0	6.43	*8.97E+07*~~8.80E+07~~
6.43	20.0	*8.56E+06*~~8.40E+06~~
Totals		*4.13E+08*~~4.06E+08~~

Table 5.2.12

CALCULATED MPC-37 PWR NEUTRON SOURCE PER ASSEMBLY
FOR 65,000 MWD/MTU BURNUP AND ~~10~~ *8* YEAR COOLING

Lower Energy (MeV)	**Upper Energy (MeV)**	**65,000 MWD/MTU ~~10~~*8*-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	*6.80E+07*~~6.30E+07~~
4.0e-01	9.0e-01	*1.48E+08*~~1.37E+08~~
9.0e-01	1.4	*1.47E+08*~~1.37E+08~~
1.4	1.85	*1.17E+08*~~1.09E+08~~
1.85	3.0	*2.18E+08*~~2.03E+08~~
3.0	6.43	*1.98E+08*~~1.84E+08~~
6.43	20.0	*1.89E+07*~~1.75E+07~~
Totals		*9.16E+08*~~8.50E+08~~*9*

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		
Region	**BPRA**	**TPD**
Upper End Fitting (kg of steel)	2.62	2.3
Upper End Fitting (kg of inconel)	0.42	0.42
Gas Plenum Spacer (kg of steel)	0.77488	1.71008
Gas Plenum Springs (kg of steel)	0.67512	1.48992
In-core (kg of steel)	13.2	N/A

Table 5.2.16 DESIGN BASIS COBALT-60 ACTIVITIES FOR BURNABLE POISON ROD ASSEMBLIES AND THIMBLE PLUG DEVICES		
Region	**BPRA**	**TPD**
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

5.4 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. All of these data are based on ENDF/B data. MCNP has been extensively benchmarked against experimental data by the large user community. References [5.4.2], [5.4.3], and [5.4.4] are three examples of the benchmarking that has been performed.

The energy distribution of the source term, as described earlier, is used explicitly in the MCNP model. A different MCNP calculation is performed for each of the three source terms (neutron, decay gamma, and ^{60}Co). The axial distribution of the fuel source term is described in Table 2.1.5 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}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^2_{total}, is the sum of the estimated variances of the individual dose rates S^2_i. 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.

$$R_i = \frac{\sqrt{S_i^2}}{T_i} \quad \text{(Equation 5.4.2)}$$

$$S_{Total}^2 = \sum_{i=1}^{n} S_i^2 \quad \text{(Equation 5.4.3)}$$

$$T_{Total} = \sum_{i=1}^{n} T_i \qquad \text{(Equation 5.4.4)}$$

$$R_{Total} = \frac{\sqrt{S^2_{Total}}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^{n} S_i^2}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^{n} (R_i \times T_i)^2}}{T_{Total}} \qquad \text{(Equation 5.4.5)}$$

where,

i	=	tally component index
n	=	total number of components
T_{Total}	=	total estimated tally
T_i	=	tally i component
S^2_{Total}	=	total estimated variance
S^2_i	=	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 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 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 ~~a~~ regionalized loading pattern*s*, as mentioned in Section 5.0 and described in Subsection ~~2.1~~*1.2*. Th*ese*~~is~~ loading pattern*s* authorize~~s~~ fuel of higher decay heat (i.e., higher burnups and shorter cooling times) to be stored in certain regions of the basket. ~~From a shielding perspective, placing the older fuel on the outside provides shielding for the inner fuel in the radial direction.~~ 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 *R*~~r~~egion ~~1~~ *1 and Region 2. Region 1*~~(which~~ contains 38% of total number of assemblies for the MPC-32 and 47% for the MPC-68~~)~~ ~~and 2~~. 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 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

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.

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])

Gamma Energy (MeV)	**(rem/hr)/ (photon/cm^2-s)**
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])	
Gamma Energy (MeV)	**(rem/hr)/ (photon/cm^2-s)**
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)†/(n/cm^2-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,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE HI-TRAC VW						
1	*711*~~612~~	*<1*~~<1~~	*489*~~458~~	*67*~~65~~	*1268*~~1136~~	*1268*~~1136~~
2	*2242*~~1930~~	*2*~~2~~	*<1*~~<1~~	*319*~~313~~	*2563*~~2246~~	*2563*~~2246~~
3	*7*~~6~~	*<1*~~<1~~	*128*~~120~~	*2*~~2~~	*139*~~129~~	*210*~~201~~
4	*15*~~13~~	*<1*~~<1~~	*227*~~212~~	*<1*~~<1~~	*244*~~227~~	*371*~~354~~
5 (bottom lid)	*465*~~395~~	*<1*~~<1~~	*1802*~~1687~~	*79*~~77~~	*2346*~~2160~~	*2346*~~2160~~
ONE METER FROM THE HI-TRAC VW						
1	*541*~~468~~	*<1*~~<1~~	*61*~~57~~	*53*~~52~~	*657*~~578~~	*657*~~578~~
2	*1141*~~985~~	*<1*~~<1~~	*5*~~5~~	*116*~~114~~	*1263*~~1105~~	*1264*~~1106~~
3	*191*~~164~~	*<1*~~<1~~	*75*~~70~~	*20*~~19~~	*286*~~254~~	*326*~~295~~
4	*8*~~6~~	*<1*~~<1~~	*127*~~119~~	*<1*~~<1~~	*137*~~127~~	*208*~~198~~
5	*259*~~221~~	*<1*~~<1~~	*985*~~923~~	*20*~~20~~	*1266*~~1165~~	*1266*~~1165~~

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}Co from the spacer grids, and ^{60}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

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE HI-TRAC VW						
1	*433*~~371~~	*1*~~<1~~	*301*~~282~~	*54*	*740*~~658~~	*740*~~658~~
2	*1266*~~1083~~	*5*~~5~~	*<1*~~<1~~	*27*~~26~~	*1298*~~1115~~	*1298*~~1115~~
3	*2*~~2~~	*<1*~~<1~~	*69*~~64~~	*<1*~~<1~~	*73*~~68~~	*111*~~106~~
4	*15*~~13~~	*<1*~~<1~~	*227*~~212~~	*<1*~~<1~~	*244*~~227~~	*371*~~354~~
5 (bottom lid)	*465*~~395~~	*<1*~~<1~~	*1802*~~1687~~	*80*~~78~~	*2347*~~2161~~	*2347*~~2161~~
ONE METER FROM THE HI-TRAC VW						
1	*294*~~253~~	*1*~~<1~~	*34*~~32~~	*44*	*333*~~290~~	*334*~~290~~
2	*657*~~566~~	*2*~~2~~	*3*~~3~~	*10*~~10~~	*671*~~581~~	*672*~~582~~
3	*95*~~82~~	*<1*~~<1~~	*41*~~38~~	*1*~~1~~	*138*~~122~~	*160*~~144~~
4	*8*~~6~~	*<1*~~<1~~	*127*~~119~~	*<1*~~<1~~	*137*~~127~~	*208*~~198~~
5	*259*~~221~~	*<1*~~<1~~	*985*~~923~~	*20*~~19~~	*1265*~~1164~~	*1265*~~1164~~

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}Co from the spacer grids, and ^{60}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,γ) Gammas (mrem/hr)**	**^{60}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}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,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW					
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
ONE METER FROM THE HI-TRAC VW					
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}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	
Dose Component	**45,000 MWD/MTU *4*.5-Year Cooling (mrem/yr)**
Fuel gammas	*15.8*~~13.1~~
^{60}Co Gammas	*2.2*~~2.0~~
Neutrons	*0.2*~~0.2~~
Total	*18.2*~~15.3~~

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, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.4.7 DOSE VALUES USED IN CALCULATING ANNUAL DOSE FROM VARIOUS HI-STORM FW ISFSI CONFIGURATIONS 45,000 MWD/MTU AND ~~5~~4.5-YEAR COOLING ZIRCALOY CLAD FUEL			
Distance	**A Side of Overpack (mrem/yr)**	**B Top of Overpack (mrem/yr)**	**C Side of Shielded Overpack (mrem/yr)**
100 meters	*396.0~~337~~*	*44.0~~37~~*	*79.2~~67~~*
200 meters	*61.7~~52~~*	*6.9~~6~~*	*12.3~~10~~*
300 meters	*16.4~~14~~*	*1.8~~2~~*	*3.3~~3~~*
400 meters	*5.34*	*0.6~~0.5~~*	*1.1~~0.9~~*
500 meters	*2.0~~1.6~~*	*0.2~~0.2~~*	*0.4~~0.3~~*
600 meters	*0.8~~0.7~~*	*0.1~~0.07~~*	*0.2~~0.1~~*

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)**
ADJACENT TO THE HI-TRAC VW		
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
ONE METER FROM THE HI-TRAC VW		
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.

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_{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 ($k_{eff} \sim 0.6$). 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}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}U and ^{236}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}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.

- 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_{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_{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[††], Tables 6.1.1, ~~and~~ 6.1.2 *and 6.1.6* list the bounding maximum k_{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_{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_{eff} values for the HI-STORM FW system are below the limiting design criteria ($k_{eff} < 0.95$) when fully flooded and loaded with any of the candidate fuel assemblies and basket configurations. Analyses for the various conditions of flooding that support the conclusion that the fully flooded condition corresponds to the highest reactivity, and thus is most limiting, are presented in Section 6.4. The capability of the HI-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 in 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.

[††] The assembly classes for BWR and PWR fuel are defined in Section 6.2.

TABLE 6.1.1

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-37 (HI-TRAC VW)

Fuel Assembly Class	4.0 wt% ^{235}U Maximum Enrichment†		5.0 wt% ^{235}U Maximum Enrichment†	
	Minimum Soluble Boron Concentration (ppm)	Maximum k_{eff}	Minimum Soluble Boron Concentration (ppm)	Maximum k_{eff}
14x14A	1000	0.8946	1500	0.8983
14x14B	1000	0.9121	1500	0.9172
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

† For maximum allowable enrichments between 4.0 wt% ^{235}U and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified 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}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

TABLE 6.1.3

REPRESENTATIVE k_{eff} VALUES FOR MPC-37 AND MPC-89 IN THE HI-STORM FW OVERPACK

MPC	Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum k_{eff}
MPC-37	17x17B	5.0	0.6076
MPC-89	10x10A	4.8	0.3986

TABLE 6.1.4

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-37
WITH UP TO 12 DFCs

Fuel Assembly Class of Undamaged Fuel	**4.0 wt% ^{235}U Maximum Enrichment for Undamaged Fuel and Damaged Fuel/Fuel Debris** †		**5.0 wt% ^{235}U Maximum Enrichment for Undamaged Fuel and Damaged Fuel/Fuel Debris** †	
	Minimum Soluble Boron Concentration (ppm)	**Maximum k_{eff}**	**Minimum Soluble Boron Concentration (ppm)**	**Maximum k_{eff}**
All 14x14, 16x16A	1300	0.9023	1800	0.9163
All 15x15, all 17x17	1800	0.9032	2300	0.9276

† For maximum allowable enrichments between 4.0 wt% ^{235}U and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified for each assembly class.

TABLE 6.1.5

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-89
WITH UP TO 16 DFCs

Fuel Assembly Class	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**	**Maximum k_{eff}**
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_{eff} VALUES FOR THE MPC-89
WITH LOW ENRICHED, CHANNELED BWR FUEL

Fuel Assembly Class	***Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)***	***Maximum k_{eff}***
All BWR Classes	*3.3*	*0.9325*

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_{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.

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 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 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_{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 are completely flooded while the remainder of the MPC is only filled with steam consisting of ordinary water at partial density (0.002 g/cm^3 or less). Assuming this condition, the case resulting in the highest maximum k_{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_{eff} in comparison with the maximum k_{eff} for the fully flooded condition. For all configurations, the preferential flooding condition results in a lower maximum k_{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_{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.

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;

⇒ σ_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

⇒ σ_B is the standard error of the bias (which includes the K multiplier for 95% probability at the 95% confidence level; see Appendix 6.A).

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), the DFCs are covered by the neutron absorber even if they were to move axially.

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 ~~a~~ *the* ~~full~~ *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.

<u>Bounding Undamaged Assemblies</u>

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. The highest maximum k_{eff} values correspond to a 16x16 array of bare fuel rods for the MPC-37, and for an 11x11 array for the MPC-89. In all cases, the maximum k_{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

- 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 cladded 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 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_{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}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_{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

MPC-37 (17x17B, 5.0% ENRICHMENT)		
Flooded Condition (% Full)	Maximum k_{eff}, Vertical Orientation	Maximum k_{eff}, Horizontal Orientation
25	0.9175	0.8306
50	0.9325	0.9093
75	0.9357	0.9349
100	0.9380	0.9380
MPC-89 (10x10A, 4.8% ENRICHMENT)		
Flooded Condition (% Full)	Maximum k_{eff}, Vertical Orientation	Maximum k_{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	**Maximum k_{eff}**	
	MPC-37 (17x17B, 5.0% ENRICHMENT)	**MPC-89 (10x10A, 4.8% ENRICHMENT)**
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† in g/cm³	**Maximum k_{eff}**						
	MPC-89 10x10A, 4.8%	**MPC-37 (1500ppm) 17x17B, 4.0 %**		**MPC-37 (2000ppm) 17x17B, 5.0 %**		**MPC-37† (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

† External moderator is modeled at 100%.

† With undamaged and damaged fuel. All other cases with undamaged fuel only

TABLE 6.4.6

MAXIMUM k_{eff} VALUES IN THE MPC-37 WITH UNDAMAGED (15x15F) AND DAMAGED FUEL

Bare Rod Array inside the DFC	**Maximum k_{eff}, 4.0 wt%**	**Maximum k_{eff}, 5.0 wt%**
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

TABLE 6.4.7

MAXIMUM k_{eff} VALUES IN THE MPC-89 WITH UNDAMAGED (10x10A) AND DAMAGED FUEL

Bare Rod Array inside the DFC	**Maximum k_{eff}, 4.8 wt% (planar average)**
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

~~Not Used~~ *MAXIMUM k_{eff} VALUES IN THE MPC-89 WITH LOW ENRICHED (3.3 wt% ^{235}U), CHANNELED, BWR FUEL*

Rod Array inside the Channel	***Maximum k_{eff}***
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*

Attachment 3 to Holtec Letter 5018019
October 13, 2011

TABLE 6.4.9

COMPARISON OF MCNP CONVERGENCE PARAMETERS

Calculation Parameters		Maximum k_{eff}	
Particles per Cycle	Skipped Cycles	**MPC-37 (17x17B, 5.0% ENRICHMENT)**	**MPC-89 (10x10A, 4.8% ENRICHMENT)**
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_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-37 WITH CONDITIONS OF FILLED AND VOIDED GUIDE AND INSTRUMENT TUBES AT 5 % ENRICHMENT

Fuel Assembly Class	Maximum k_{eff}, Filled Tubes	Maximum k_{eff}, Voided Tubes
14x14A	0.8983	0.8887
14x14B	0.9172	0.9015
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

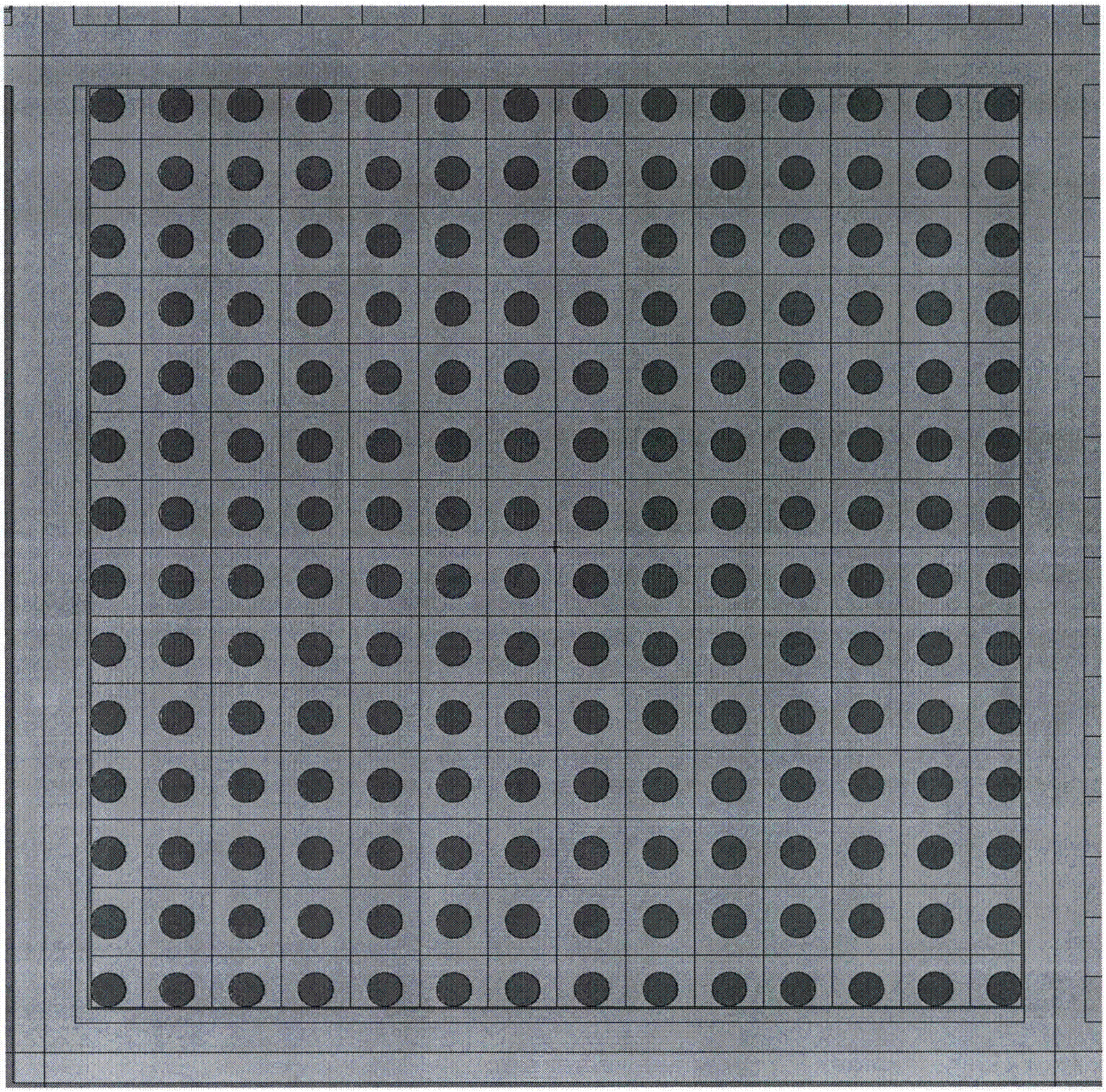


Figure 6.4.1: Calculational Model (planar cross-section) of a DFC in a MPC-37 cell with a 14x14 array of bare fuel rods

CHAPTER 11: RADIATION PROTECTION†

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.

† 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 region s 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.

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

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	The dose from a HI-STORM FW storage system is minimized because of the following advantages: 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. b. Tight packing of overpacks on the ISFSI (that maximizes self-shielding) is possible because a large spacing between the modules is not necessary.	A,B
9.	MPC, HI-TRAC VW	The occupational dose from loading a HI-STORM FW overpack is minimized because of: a. A well-shielded HI-TRAC VW transfer cask. b. Regionalized fuel loading. c. A short water draining time (less than 2 hours) for the MPC. d. Reduced overall MPC welding time because the welding machine does not have to be removed and replaced to weld the secondary lid. e. Reduced time and personnel needed to install the MPC in the HI-STORM FW overpack due to vertical (gravity-aided) insertion. f. Reduced drying time because of use of porosity-free Metamic-HT.	B
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

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**
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

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.

Attachment 3 to Holtec Letter 5018019
October 13, 2011

	Table 11.3.1 ASSUMED PARAMETERS FOR DOSE ESTIMATE UNDER SHORT-TERM OPERATIONS AND UNDER LONG-TEM STORAGE	
	Item	**Value**
1.	**MPC-Contents (MPC-37)[1]**	**45,000 MWD/MTU and 4.5 years**
2.	**Weight of HI-TRAC VW Full of Fuel and Water**	**125 tons**
3.	**HI-STORM Concrete Density**	**150 lb/cubic feet**

[1] The case of MPC-37 is used but similar results are expected for the MPC-89.

TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM				
Task Description (See Chapter 9 for detailed description of operations)	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Average Dose Rate at worker location (mrem/hr)**	**Exposure (mrem)**
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*~~1~~	*40.0*~~40~~
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*~~50.4~~	*83.5*~~75.6~~
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*~~50.4~~	*34.3*~~31.1~~

TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM				
Task Description (See Chapter 9 for detailed description of operations)	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Average Dose Rate at worker location (mrem/hr)**	**Exposure (mrem)**
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*~~160.3~~	*198.7*~~181.7~~
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*~~218.2~~	*61.2*~~58.2~~
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	0.0

TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM				
Task Description (See Chapter 9 for detailed description of operations)	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Average Dose Rate at worker location (mrem/hr)**	**Exposure (mrem)**
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*~~130~~	*88.8*~~78.0~~
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*~~31~~	*205.2*~~170.5~~
TOTAL EXPOSURE (person-mrem)	*711.6*~~635.0~~			

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 ~~was~~ *is* the minimum distance analyzed in Chapter ~~55. Table 11.4.1 provides the annual dose results for a single HI-STORM FW under the conditions specified in Table 11.3.1~~. 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 indicate that the calculated annual dose is less than the regulatory limit of 25 mrem/year at a distance of 300 meters for a single cask and at 500 meters for the cask group of HI-STORM FW systems containing Table 11.3.1 fuel.~~ 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 analysis 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 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.