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June 6, 2016

Yen-Ju Chen, Sr. Project Manager – Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety and Safeguards

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

Docket No. 72-1014, Certificate of Compliance (CoC) No. 1014

Subject: Transmittal of RSI Responses Supporting HI-STORM 100 LAR 1014-11

Reference: [1] "Acceptance Review of Request for Amendment No. 11 To Certificate of Compliance No. 1014 for the HI-STORM 100 Multipurpose Canister Storage System – Request for Supplemental Information" (Letter from Yen-Ju Chen (NRC) to Kimberly Manzione (Holtec) dated May 11, 2016)
[2] "Modification of Requested Changes on HI-STORM 100 Amendment 11 Request" (Letter from Kimberly Manzione (Holtec) to Yen Chen (NRC) dated April 22, 2016)
[3] "Supporting Information for License Amendment Request 11 (1014-11) to the HI-STORM 100 CoC" (Letter from Royston Ngwayah (Holtec) to Jose Cuadrado (NRC) dated February 16, 2016)
[4] "Holtec International HI-STORM 100 Multipurpose Canister Storage System Amendment Request 1014-11" (Letter from Royston Ngwayah (Holtec) to Mark Lombard (NRC) dated January 29, 2016)

Dear Ms. Chen:

This letter is in response to the staff's request for supplemental information regarding HI-STORM 100 LAR 1014-11 [1]. Also requested in [1] is resubmittal of changes to the HI-STORM 100 FSAR, excluding Proposed Changes 2 and 3 in [2, 4]. This letter contains an updated version of HI-STORM 100 Amendment 11, including responses to the RSIs. Holtec

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hereby submits the following documents in response to [1]: scope of proposed changes (Enclosure 1), CoC (Enclosure 2), CoC Appendix A (Enclosure 3), CoC Appendix B (Enclosure 4), proposed HI-STORM 100 FSAR changes (Enclosure 5) and responses to request for additional information (Enclosure 6).

The changes identified in the enclosures of this letter encompass changes from the submitted amendment request [4], responses to the RSIs, and minor editorial FSAR text changes to clarify items in the original submittal [1]. Calculation packages [3] submitted in support of the initial amendment request are applicable to the changes in the enclosures to this letter.

If you have any questions please contact me at 856-797-0900 ext. 3844.

Sincerely,

Royston Ngwayah
Licensing Engineer,
Holtec International

cc: (via email)
Mark Lombard (USNRC)
Steve Ruffin (USNRC)
Bo Pham (NRC)

Enclosures:
Enclosure 1: Scope of Proposed Changes for HI-STORM 100 LAR 1014-11
Enclosure 2: Proposed CoC 1014 Amendment 11
Enclosure 3: Proposed CoC 1014 Amendment 11 Appendix A
Enclosure 4: Proposed CoC 1014 Amendment 11 Appendix B
Enclosure 5: HI-STORM 100 FSAR Proposed Changed pages
Enclosure 6: Responses to Request for Supplemental Information

Enclosure 1 to Holtec Letter 5014810

LAR 1014-11, REVISION 0

SUMMARY OF PROPOSED CHANGES

Proposed Change #1

Increase the per storage location weight limit for cells authorized for DFCs in the MPC-68, 68FF and MPC-68M in the HI-STORM 100 System CoC, Appendix B, Table 2.1-1.

Reason for Proposed Change #1

This proposed change allows storage of additional fuel types in the HI-STORM 100 System.

Justification for Proposed Change #1

The new fuel weights have been structurally evaluated. The overall MPC bounding weight remains unchanged, and therefore the current FSAR analyses remain bounding. A marked copy of FSAR Chapters 2 and 3 are provided to show the changes.

Proposed Change # 2

It is proposed to provide more options on surveillance requirements and actions to be taken in the event of Overpack vents blockage when containing a loaded MPC.

Reason for Proposed Change # 2

To relax surveillance requirements for MPCs with much lower heat loads than design basis maximum. This will help reduce the necessary surveillance burden on users of ISFSI sites prone to unfavorable weather conditions periodically.

Justification for Proposed Change # 2

For heat loads below an established threshold, during OVERPACK vents blockages that render the system inoperable, thermal analysis has shown that the accident condition temperature limits for system components and the fuel cladding are not exceeded under steady state conditions. Mark-ups of FSAR Chapters 4 and 12 and CoC Appendix A are provided with the identified changes.

Proposed Change 3

It is proposed to evaluate mixture of low enriched CILC fuel and normal fuel.

Reason for Proposed Change #3

This change provides flexibility to system users by permitting the loading of low enriched CILC fuel with other undamaged fuel assemblies. Therefore, the inventory of fuel assemblies in the

spent fuel pool is reduced and complex activities (i.e. insertion of fuel assemblies into DFCs) are avoided.

Justification for Proposed Change #3

Criticality analysis performed in support of this change indicates that k_{eff} remains below 0.95 limit under all analyzed storage conditions. Mark-ups of FSAR Chapters 2 and 6 and CoC Appendix B are provided to show the changes.

Proposed Change #4

It is proposed to increase the enrichment limit for 10x10G (BWR) fuel assembly from 4.6 wt.% ^{235}U to 4.75 wt.% ^{235}U.

Reason for Proposed Change #4

Expand the range of allowable contents to include assemblies in use.

Justification for Proposed Change #4

Criticality analysis performed indicates that k_{eff} remains below 0.95 limit under all analyzed storage conditions. Mark-ups of Chapters 2 and 6 and CoC Appendix B are provided to show the changes.

Proposed Change #5

It is proposed to add new minimum soluble boron concentration limits for the 17x17A (PWR) fuel assemblies an MPC-32.

Reason for Proposed Change #5

Enhance practicality of loading operations and reduce undue burden on users.

Justification for Proposed Change #5

The 17x17A fuel assemblies have lower uranium weight than 17x17B/C fuel assemblies, therefore a different soluble boron concentration limit is applied for loading of 17x17A fuel assemblies. Criticality analysis performed indicates that for the proposed lower soluble boron concentration k_{eff} limit is below 0.95 for all conditions analyzed. Mark-ups of Chapters 2 and 6 and CoC Appendix A are provided to show the changes.

SUMMARY OF PROPOSED CHANGES

Proposed Change #6

It is proposed to increase the burnup limit to accommodate non-fuel hardware (NFH) consisting of NSA in combination with other control components.

Reason for Proposed Change #6

Permit loading of NFH in combination with other control components.

Justification for Proposed Change #6

In accordance with the shielding analysis, to maintain Co-60 activity at or below 895 Ci, the cooling time was increased to accommodate the increased burnup. Mark-ups of FSAR Chapter 2 and CoC Appendix B are provided to identify the changes.

Proposed Changes #7

It is proposed to add thoria rods/canister as contents for the MPC-68M.

Reason for Proposed Change #7

Expand the allowable contents for the MPC-68M to include the thoria rods/canister, which are already approved for other MPC-68 models.

Justification for Proposed Change #7

Shielding analysis indicate the difference in dose rates between the MPC-68 and MPC-68M, when any other content is loaded with the thoria rod canister, is very small. Therefore the thoria rod canister is acceptable for loading in the MPC-68M. Criticality analysis concludes that the case of the thoria rod canister in MPC-68M is bounded by the analysis for the MPC-68 and MPC-68F, and therefore the thoria rod canister is permissible for loading in the MPC-68M together with any approved content.

Proposed Changes #8

It is proposed to add a second permissible composition for thoria rods for all MPC-68 models.

Reason for Proposed Change #8

Enclosure 1 to Holtec Letter 5014810

LAR 1014-11, REVISION 0

SUMMARY OF PROPOSED CHANGES

To expand the approved composition of thoria rods in the CoC to include thoria rods composition submitted to the NRC via an exemption request for Dresden Nuclear Power Station (NRC Docket No. 72-37, TAC No. L24989).

Justification for Proposed Change #8

The new composition of thoria rods has been shown by analysis to have a negligible impact on dose rates and criticality margins. Mark-ups of FSAR Chapters 2, 5 and 6 and CoC Appendix B CoC identify the changes.

Clarifications and Editorial Suggestions in the CoC/FSAR

A) CoC Appendix B (Section 3.4), single failure proof criteria and definition have been clarified in accordance with the HI-STORM FW (NRC Docket No. 72-1032) System

B) CoC Appendix B (Section 3.3.2), has been modified to include ASME Section II with the ASME Section III Code Alternative submittals.

C) CoC Appendix B, Section 2.4.3 has been modified to remove the burnup calculation. General licensees are responsible for ensuring that they comply with the requirements in the CoC for heat load, burnup, and enrichment. The calculation is only one method of performing this evaluation. Newer submittals for the HI-STORM FW and HI-STORM UMAX (NRC Docket No. 72-1040) do not contain the specific equation.

D) CoC Appendix A, definition of Repaired/Reconstituted Fuel Assembly has been modified to clarify that if dummy stainless steel rods are present in the loaded spent fuel assemblies, the dummy/replacement rods will be considered in the site-specific dose calculations.

E) CoC Appendix A, Table 3-1 has been updated to clarify that the allowable heat loads in Notes 5 and 6 apply to vacuum drying and FHD respectively.

F) CoC Appendix A, Tables 3-3 and 3-4 have been modified to clarify that the heat load limits for regionalized and uniform loadings for MPCs-68/68F/68FF as provided in both tables, are also applicable to MPC-68M. This aligns the CoC with analysis previously performed and provided in the FSAR.

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.
1014	05/31/00	05/31/20	72-1014	1~~0~~1	TBD	USA/72-1014

Issued To: (Name/Address)

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

Safety Analysis Report Title

Holtec International Inc.,
Final Safety Analysis Report for the
HI-STORM 100 Cask System

CONDITIONS

This certificate is conditional 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) for aboveground systems or the attached Appendix A-100U (Technical Specifications) and Appendix B-100U (Approved Contents and Design Features) for underground systems, and the conditions specified below:

1. CASK

   a. Model No.: HI-STORM 100 Cask System

   The HI-STORM 100 Cask System (the cask) consists of the following components: (1) interchangeable multi-purpose canisters (MPCs), which contain the fuel; (2) a storage overpack (HI-STORM), which contains the MPC during storage; and (3) a transfer cask (HI-TRAC), which contains the MPC during loading, unloading and transfer operations. The cask stores up to 32 pressurized water reactor fuel assemblies or 68 boiling water reactor fuel assemblies.

   b. Description

   The HI-STORM 100 Cask 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 cask comprises three discrete components: the MPC, the HI-TRAC transfer cask, and the HI-STORM storage overpack.

   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 such as the neutron absorbers. 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, which contains neutron absorbing material, provides criticality control.

1. b. Description (continued)

There are nine types of MPCs: the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, MPC-68F, MPC-68FF, and MPC-68M. The number suffix indicates the maximum number of fuel assemblies permitted to be loaded in the MPC. All nine MPC models have the same external diameter.

The HI-TRAC transfer cask provides shielding and structural protection of the MPC during loading, unloading, and movement of the MPC from the spent fuel pool 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. Two sizes of HI-TRAC transfer casks are available: the 125 ton HI-TRAC and the 100 ton HI-TRAC. The weight designation indicates the approximate weight of a loaded transfer cask during any loading, unloading, or transfer operation. Both transfer cask sizes have identical cavity diameters. The 125 ton HI-TRAC transfer cask has thicker shielding and larger outer dimensions than the 100 ton HI-TRAC transfer cask.

Above Ground Systems

The HI-STORM 100 or 100S storage overpack provides shielding and structural protection of the MPC during storage. The HI-STORM 100S is a variation of the HI-STORM 100 overpack design that includes a modified lid which incorporates the air outlet ducts into the lid, allowing the overpack body to be shortened. 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 four air inlets at the bottom and four air outlets at the top to allow air to circulate naturally through the cavity to cool the MPC inside. The inner shell has supports attached to its interior surface to guide the MPC during insertion and removal, provide a medium to absorb impact loads, and allow cooling air to circulate through the overpack. A loaded MPC is stored within the HI-STORM 100 or 100S storage overpack in a vertical orientation. The HI-STORM 100A and 100SA are variants of the HI-STORM 100 family and are outfitted with an extended baseplate and gussets to enable the overpack to be anchored to the concrete storage pad in high seismic applications.

Underground Systems

The HI-STORM 100U System is an underground storage system identified with the HI-STORM 100 Cask System. The HI-STORM 100U storage Vertical Ventilated Module (VVM) utilizes a storage design identified as an air-cooled vault or caisson. The HI-STORM 100U storage VVM relies on vertical ventilation instead of conduction through the soil, as it is essentially a below-grade storage cavity. Air inlets and outlets allow air to circulate naturally through the cavity to cool the MPC inside. The subterranean steel structure is seal welded to prevent ingress of any groundwater from the surrounding subgrade, and it is mounted on a stiff foundation. The surrounding subgrade and a top surface pad provide significant radiation shielding. A loaded MPC is stored within the HI-STORM 100U storage VVM in the vertical orientation.

2. OPERATING PROCEDURES

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance. The user's site-specific written operating procedures shall be consistent with the technical basis described in Chapter 8 of the FSAR.

3. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Written cask acceptance tests and maintenance program shall be prepared consistent with the technical basis described in Chapter 9 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. This test shall include the base metals of the MPC shell and baseplate. A helium leak test shall also be performed on the base metal of the fabricated MPC lid. In the field, a helium leak test shall be performed on the vent and drain port confinement welds and cover plate base metal. The confinement boundary leakage rate tests shall be performed in accordance with ANSI N14.5 to "leaktight" criteria. 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 requirements. Re-testing shall be performed until the leakage rate acceptance criterion is met.

4. 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 cask system.

5. HEAVY LOADS REQUIREMENTS

Each lift of an MPC, a HI-TRAC transfer cask, or any HI-STORM 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 (under 10 CFR 50.59 or 10 CFR 72.48, if 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 5.5 of Appendix A and Sections 3.4.6 and 3.5 (if applicable) of Appendix B, for above ground systems, section 5.5 of Appendix A-100U for the underground systems.

6. APPROVED CONTENTS

Contents of the HI-STORM 100 Cask System must meet the fuel specifications given in Appendices B for aboveground systems or B-100U for underground systems to this certificate.

7. DESIGN FEATURES

Features or characteristics for the site, cask or ancillary equipment must be in accordance with Appendices B for aboveground systems or B-100U for underground systems to this certificate.

8. CHANGES TO THE CERTIFICATE OF COMPLIANCE

The holder of this certificate who desires to make changes to the certificate, which includes Appendices A and A-100U (Technical Specifications) and Appendices B and B-100U (Approved Contents and Design Features), shall submit an application for amendment of the certificate.

9. SPECIAL REQUIREMENTS FOR FIRST SYSTEMS IN PLACE

a. For the storage configuration, each user of a HI-STORM 100 Cask and HI-STORM 100U Cask with a heat load equal to or greater than 20 kW shall perform a thermal validation test in which the user measures the total air mass flow rate through the cask system using direct measurements of air velocity in the inlet vents. The user shall then perform an analysis of the cask with the taken measurements to demonstrate that the measurements validate the analytic methods described in Chapter 4 of the FSAR. The thermal validation test and analysis results shall be submitted in a letter report to the NRC pursuant to 10 CFR 72.4 within 180 days of the user's loading of the first cask with heat load equal to or greater than 20 kW. To satisfy condition 9(a) for casks of the same system type (i.e., HI-STORM 100 casks, HI-STORM 100U casks), in lieu of additional submittals pursuant to 10 CFR 72.4, users may document in their 72.212 report a previously performed test and analysis submitted by letter report to the NRC that demonstrates validation of the analytic methods described in Chapter 4 of the FSAR.

b. For transfer configuration, each user of the HI-STORM 100 Cask and HI-STORM 100U Cask shall procure, if necessary, a Supplemental Cooling System (SCS) capable of providing the thermal-hydraulic characteristics (coolant temperature at the annulus inlet, coolant temperature located at the annulus outlet, and coolant flow rate) that will ensure that thermal limits (described in Appendix 2.C of the FSAR) are not exceeded during transfer operations. The thermal-hydraulic characteristics of the SCS shall be determined using the analytical methods described in Chapter 4 for the transfer configuration. For the transfer configuration, each first time user shall measure the SCS thermal-hydraulic characteristics to validate the performance of the SCS. The SCS analysis and validation shall be documented in an update to the 72.212 report within 180 days of the user's first transfer operation with the SCS. Condition 9(b) does not apply to the MPC-68M.

10. PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

A dry run training exercise of the loading, closure, handling, unloading, and transfer of the HI-STORM 100 Cask 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 100 Cask 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. Operation of the HI-STORM 100 SCS or equivalent system, if applicable.

h. Transfer cask upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling arrangement.

i. Transfer of the MPC from the transfer cask to the overpack/VVM.

j. Placement of the HI-STORM 100 Cask System at the ISFSI, for aboveground systems only.

k. HI-STORM 100 Cask System unloading, including flooding MPC cavity, removing MPC lid welds. (A mockup may be used for this dry-run exercise.)

11. The NRC has approved an exemption request by the CoC applicant from the requirements of 10 CFR 72.236(f), to allow a Supplemental Cooling System to provide for decay heat removal in accordance with Section 3.1.4 of Appendices A and A-100U.

NRC FORM 651
(3-1999)
10 CFR 72

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS
Supplemental Sheet

Certificate No. 1014
Amendment No. 11~~0~~

12. AUTHORIZATION

The HI-STORM 100 Cask 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, B, A-100U, and B-100U, as applicable. The HI-STORM 100 Cask System may be fabricated and used in accordance with any approved amendment to CoC No. 1014 listed in 10 CFR 72.214. Each of the licensed HI-STORM 100 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.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

TBD, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards
Washington, DC 20555

Dated TBD

Attachments:
1. Appendix A
2. Appendix B
3. Appendix A-100U
4. Appendix B-100U

CERTIFICATE OF COMPLIANCE NO. 1014

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR THE HI-STORM 100 CASK SYSTEM

1.1 Definitions (continued)

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.
MINIMUM ENRICHMENT	MINIMUM ENRICHMENT is the minimum assembly average enrichment. Natural uranium and low enrichment blankets are not considered in determining minimum enrichment.
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 VVM (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 assemblies, instrument tube tie rods (ITTRs), vibration suppressor inserts, and components of these devices such as individual rods.

(continued)

1.1 Definitions (continued)

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/RECONSTITUTED FUEL ASSEMBLY	Spent nuclear fuel assembly which contains dummy fuel rod(s) that displaces an amount of water greater than or equal to the original fuel rod(s) and/or which contains structural repairs so it can be handled by normal means. If irradiated dummy stainless steel rods are present in the fuel assembly, the dummy/replacement rods will be considered in the site specific dose calculations.
SPENT FUEL STORAGE CASKS (SFSCs)	SFSCs are containers approved for the storage of spent fuel assemblies at the ISFSI. The HI-STORM 100 SFSC System consists of the OVERPACK/VVM 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 to transfer the MPC to or from the OVERPACK/VVM. The HI-STORM 100 System employs either the 125-Ton or the 100-Ton HI-TRAC TRANSFER CASK.

(continued)

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 and outlet vent areas are unblocked and available for flow or when air temperature requirements are met. If surveillance shows partial blockage (≤ 50%) of the duct areas, the blockage should be removed.

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~~Deleted.	A.1 ~~Remove blockage~~Deleted.	~~N/A~~Deleted.
B. SFSC Heat Removal System inoperable.	B.1 Restore SFSC Heat Removal System to operable status.	8 hours (Overpacks containing MPCs with heat loads greater than 19 kW at time of entering condition) OR 24 hours (Overpacks containing MPCs with heat loads less than or equal to 19 kW at time of entering condition)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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.	64 hours (Storage cell heat loads≤ Tables 3-3 or 3-4 limits) 24 hours (Storage cell heat loads> Tables 3-3 or 3-4 limits)
	<u>OR</u>	
	C.2.2 Transfer the MPC into a TRANSFER CASK.	64 hours (Storage cell heat loads≤ Tables 3-3 or 3-4 limits) 24 hours (Storage cell heat loads> Tables 3-3 or 3-4 limits)

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 (Overpacks containing MPCs with heat loads greater than 19 kW at time of inspection) *OR* *30 days* (Overpacks containing MPCs with heat loads less than or equal to 19 kW at time of inspection)
	OR For OVERPACKS with installed temperature monitoring equipment, verify that the difference between the average OVERPACK air outlet temperature and ISFSI ambient temperature is $\leq$ 155°F for OVERPACKS containing PWR MPCs, $\leq$ 137°F for OVERPACKS containing BWR MPCs.	24 hours (Overpacks containing MPCs with heat loads greater than 19 kW at time of inspection) *OR* *30 days* (Overpacks containing MPCs with heat loads less than or equal to 19 kW at time of inspection)

3.3 SFSC CRITICALITY CONTROL

3.3.1 Boron Concentration

LCO 3.3.1 As required by CoC Appendix B, Table 2.1-2, 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 and classification to be stored in the MPC:

a. MPC-24 with one or more fuel assemblies having an initial enrichment greater than the value in Table 2.1-2 for no soluble boron credit and $\leq$ 5.0 wt% ^{235}U: $\geq$ 400 ppmb

b. MPC-24E or MPC-24EF (all INTACT FUEL ASSEMBLIES) with one or more fuel assemblies having an initial enrichment greater than the value in Table 2.1-2 for no soluble boron credit and $\leq$ 5.0 wt% ^{235}U: $\geq$ 300 ppmb

c. Deleted.
d. Deleted.

e. MPC-24E or MPC-24EF (one or more DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS) with one or more fuel assemblies having an initial enrichment > 4.0 wt% ^{235}U and $\leq$ 5.0 wt% ^{235}U: $\geq$ 600 ppmb

f. MPC-32/32F: Minimum soluble boron concentration as required by the table below†.

Array/Class	All INTACT FUEL ASSEMBLIES		One or more DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS	
	Maximum Initial Enrichment $\leq$ 4.1 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}U (ppmb)	Maximum Initial Enrichment $\leq$ 4.1 wt% ^{235}U (ppmb)	Maximum Initial Enrichment 5.0 wt% ^{235}U (ppmb)
14x14A/B/C/D/E	1,300	1,900	1,500	2,300
15x15A/B/C/G/I	1,800	2,500	1,900	2,700
15x15D/E/F/H	1,900	2,600	2,100	2,900
16x16A/B/C	1,400	2,000	1,500	2,300
17x17A	1,600	2,200	1,800	2,600
17x17~~A/~~B/C	1,900	2,600	2,100	2,900

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

Table 3-1
MPC Cavity Drying Limits for all MPC Types

Fuel Burnup (MWD/MTU)	MPC Heat Load (kW)	Method of Moisture Removal (Notes 1 and 2)
All Assemblies ≤ 45,000	≤ 30~~[Note 5]~~ (MPC-24/24E/24EF, MPC-32/32F, MPC-68/68F/68FF) ≤ 36.9~~[Note 6]~~ (MPC-68M)	VDS[Note 5] or FHD[Note 6]
All Assemblies ≤ 45,000	> 30~~[Note 6]~~ (MPC-24/24E/24EF, MPC-32/32F, MPC-68/68F/68FF)	FHD[Note 6]
One or more assemblies > 45,000	≤ 29 (MPC-68M)	VDS[Note 4] or FHD[Note 6]
One or more assemblies > 45,000	≤ 36.9~~[Note 6]~~ (MPC-24/24E/24EF/MPC-32/32F/MPC-68/68F/68FF/MPC-68M	FHD[Note 6]

Notes:

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

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. Deleted

4. The maximum allowable decay heat per fuel storage location is 0.426 kW.

5. Maximum allowable storage cell heat load is 1.25 kW (MPC-24/24E/24EF), 0.937 kW (MPC-32/32F) and 0.441 kW (MPC-68/68F/68FF).

6. Maximum per assembly allowable heat loads under uniform or regionalized storage defined in Appendix B, Section 2.4.1 or 2.4.2.

Table 3-3: Regionalized Storage[Note 2] Cell Heat Load Limits

MPC Type	Number ofCells in Inner Region[Note 1]	Storage Cell Heat Load (Inner Region) (kW)	Number of Cells in Outer Region[Note 1]	Storage Cell Heat Load (Outer Region) (kW)
MPC-24	4	1.470	20	0.900
MPC-24E/EF	4	1.540	20	0.900
MPC-32/32F	12	1.131	20	0.600
MPC-68/68F/68FF/68M	32	0.500	36	0.275

Note 1: The location of MPC-32 and MPC-68 inner and outer region cells are defined in Appendix B Figures 2.1-3 and 2.1-4 respectively.
The MPC-24 and MPC-24E/EF cell locations are defined below:
Inner Region Cell numbers 9, 10, 15, 16 in Appendix B Figures 2.1-1 and 2.1-2 respectively.
Outer Region Cell numbers 1-8, 11-14, 17-24 in Appendix B Figures 2.1-1 and 2.1-2 respectively.

Note 2: The storage cell regionalization is defined in Note 1 in accordance with safety analyses under the heat load limits of this Table.

Table 3-4: Uniform Storage Cell Heat Load Limits

MPC Type	Heat Load (kW)
MPC-24	1.157
MPC-24E/EF	1.173
MPC-68/68F/68FF/68M	0.414
MPC-32	0.898

5.5 Cask Transport Evaluation Program

This program provides a means for evaluating various transport configurations and transport route conditions to ensure that the design basis drop limits are met. For lifting of the loaded TRANSFER CASK or OVERPACK using devices which are integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply. This program is not applicable when the TRANSFER CASK or OVERPACK is in the FUEL BUILDING or is being handled by a device providing support from underneath (i.e., on a rail car, heavy haul trailer, air pads, etc...) ~~or is being handled by a device designed in accordance with the increased safety factors of ANSI N14.6 and having redundant drop protection~~.

Pursuant to 10 CFR 72.212, this program shall evaluate the site-specific transport route conditions.

a. For free-standing OVERPACKS and the TRANSFER CASK, the following requirements apply:

1. The lift height above the transport route surface(s) shall not exceed the limits in Table 5-1 except as provided for in Specification 5.5.a.2. Also, if applying the limits in Table 5-1, the program shall ensure that the transport route conditions (i.e., surface hardness and pad thickness) are equivalent to or less limiting than either Set A or Set B in HI-STORM FSAR Table 2.2.9.

2. The program may determine lift heights by analysis based on the site-specific conditions to ensure that the impact loading due to design basis drop events does not exceed 45 g's at the top of the MPC fuel basket. These alternative analyses shall be commensurate with the drop analyses described in the Final Safety Analysis Report for the HI-STORM 100 Cask System. The program shall ensure that these alternative analyses are documented and controlled.

(continued)

5.5 Cask Transport Evaluation Program (continued)

3. The TRANSFER CASK or OVERPACK, when loaded with spent fuel, may be lifted to any height necessary during TRANSPORT OPERATIONS, provided the lifting device is designed in accordance with applicable stress limits from ANSI N14.6, and/or NUREG-0612, and has redundant drop protection features.

4. The TRANSFER CASK and MPC, when loaded with spent fuel, may be lifted to those heights necessary to perform cask handling operations, including MPC TRANSFER, provided the lifts are made with structures and components designed in accordance with the criteria specified in Section 3.5 of Appendix B to Certificate of Compliance No. 1014, as applicable.

b. For the transport of OVERPACKS to be anchored to the ISFSI pad, the following requirements apply:

1. Except as provided in 5.5.b.2, user shall determine allowable OVERPACK lift height limit(s) above the transport route surface(s) based on site-specific transport route conditions. The lift heights shall be determined by evaluation or analysis, based on limiting the design basis cask deceleration during a postulated drop event to $\leq$ 45 g's at the top of the MPC fuel basket. Evaluations and/or analyses shall be performed using methodologies consistent with those in the HI-STORM 100 FSAR.

2. The OVERPACK, when loaded with spent fuel, may be lifted to any height necessary during TRANSPORT OPERATIONS provided the lifting device is designed in accordance with applicable stress limits from ANSI N14.6, and/or NUREG-0612, and has redundant drop protection features.

(continued)

CERTIFICATE OF COMPLIANCE NO. 1014

APPENDIX B

APPROVED CONTENTS AND DESIGN FEATURES

FOR THE HI-STORM 100 CASK SYSTEM

TABLE OF CONTENTS

2.0 Approved Contents

2.1 Fuel Specifications and Loading Conditions (cont'd)

2.1.3 Regionalized Fuel Loading

Users may choose to store fuel using regionalized loading in lieu of uniform loading to allow higher heat emitting fuel assemblies to be stored than would otherwise be able to be stored using uniform loading. Regionalized loading is limited to INTACT FUEL ASSEMBLIES or UNDAMAGED FUEL ASSEMBLIES with ZR cladding. Figures 2.1-1 through 2.1-4 define the regions for the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, MPC-68FF, and MPC-68M models, respectively[1]. Fuel assembly ~~burnup,~~ decay heat~~, and cooling time~~ limits for regionalized loading are specified in Section 2.4.2. Fuel assemblies used in regionalized loading 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.

2.3 Not Used

[1] These figures are only intended to distinguish the fuel loading regions. Other details of the basket design are illustrative and may not reflect the actual basket design details. The design drawings should be consulted for basket design details.

Table 2.1-1 (page 1 of ~~29~~30)
Fuel Assembly Limits

I. MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR INTACT FUEL ASSEMBLIES listed in Table 2.1-2, with or without NON-FUEL HARDWARE and meeting the following specifications (Note 1):

a. Cladding Type:	ZR or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class.
b. Initial Enrichment:	As specified in Table 2.1-2 for the applicable fuel assembly array/class.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D,14x14E, and 15x15G	Cooling time ≥ 8 years and an average burnup ≤ 40,000 MWD/MTU.
ii. All Other Array/Classes	Cooling time ≥ 3 years and average burnup ≤ 68,200 MWD/MTU~~as specified in Section 2.4~~.
ii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 9 of ~~29~~30)
Fuel Assembly Limits

II. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

7. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:

a. Cladding Type:	ZR
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U OR 98.5 wt.% ThO_2, 1.5 wt.% UO_2 with an enrichment of 93.5 wt.% ^{235}U
c. Number of Rods Per Thoria Rod Canister:	≤ 18
d. Decay Heat Per Thoria Rod Canister:	≤ 115 Watts
e. Post-irradiation Fuel Cooling Time and Average Burnup Per Thoria Rod Canister:	A fuel post-irradiation cooling time ≥ 18 years and an average burnup ≤ 16,000 MWD/MTIHM.
f. Initial Heavy Metal Weight:	≤ 27 kg/canister
g. Fuel Cladding O.D.:	≥ 0.412 inches
h. Fuel Cladding I.D.:	≤ 0.362 inches
i. Fuel Pellet O.D.:	≤ 0.358 inches
j. Active Fuel Length:	≤ 111 inches
k. Canister Weight:	≤ 550 lbs, including fuel

III. MPC MODEL: MPC-68 and MPC-68FF

A. Allowable Contents

1. Uranium oxide or MOX BWR INTACT FUEL ASSEMBLIES listed in Table 2.1-3, with or without channels and meeting the following specifications:

a. Cladding Type:	ZR or Stainless Steel (SS) as specified in Table 2.1-3 for the applicable fuel assembly array/class
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly	
i. Array/Classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A	Cooling time ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU (or MWD/MTIHM).
ii. Array/Class 8x8F	Cooling time ≥ 10 years and an average burnup ≤ 27,500 MWD/MTU.
iii. Array/Classes 10x10D and 10x10E	Cooling time ≥ 10 years and an average burnup ≤ 22,500 MWD/MTU.
iv. All Other Array/Classes	Cooling time ≥ 3 years and an average burnup ≤ 65,000 MWD/MTU~~As specified in Section 2.4~~.

III. MPC MODEL: MPC-68 and MPC-68FF (continued)

A. Allowable Contents (continued)

2. Uranium oxide or MOX BWR DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, with or without channels, placed in DAMAGED FUEL CONTAINERS. Uranium oxide and MOX BWR DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS shall meet the criteria specified in Table 2.1-3, and meet the following specifications:

a. Cladding Type:	ZR or Stainless Steel (SS) in accordance with Table 2.1-3 for the applicable fuel assembly array/class.
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	
i. Array/Classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A.	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
ii. All Other Array Classes	≤ 4.0 wt.% ^{235}U.
c. Initial Maximum Rod Enrichment	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	Cooling time ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU (or MWD/MTIHM).
ii. Array/Class 8x8F	Cooling time ≥ 10 years and an average burnup ≤ 27,500 MWD/MTU.
iii. Array/Class 10x10D and 10x10E	Cooling time ≥ 10 years and an average burnup ≤ 22,500 MWD/MTU.
iv. All Other Array/Classes	Cooling time ≥ 3 years and an average burnup ≤ 65,000 MWD/MTU~~As specified in Section 2.4~~.

Table 2.1-1 (page 14 of ~~29~~30)
Fuel Assembly Limits

III. MPC MODEL: MPC-68 and MPC-68FF (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly

i. Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	≤ 115 Watts
ii. Array/Class 8x8F	≤ 183.5 Watts
iii. Array/Classes 10x10D and 10x10E	≤ 95 Watts
iv. All Other Array/Classes	As specified in Section 2.4.

f. Fuel Assembly Length

i. Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	≤ 135.0 inches (nominal design)
ii. All Other Array/Classes	≤ 176.5 inches (nominal design)

g. Fuel Assembly Width

i. Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	≤ 4.70 inches (nominal design)
ii. All Other Array/Classes	≤ 5.85 inches (nominal design)

h. Fuel Assembly Weight

i. Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	≤ 550 lbs, including channels and DFC
ii. All Other Array/Classes	≤ ~~7~~830 lbs, including channels and DFC

III. MPC MODEL: MPC-68 and MPC-68FF (continued)

A. Allowable Contents (continued)

3. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:

a. Cladding type	ZR
b. Composition	98.2 wt.% ThO_2, 1.8 wt.% UO_2 with an enrichment of 93.5 wt.% ^{235}U OR 98.5 wt.% ThO_2, 1.5 wt.% UO_2 with an enrichment of 93.5% wt.% ^{235}U~~.~~
c. Number of Rods per Thoria Rod Canister:	≤ 18
d. Decay Heat Per Thoria Rod Canister:	≤ 115 Watts
e. Post-irradiation Fuel Cooling Time and Average Burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time ≥ 18 years and an average burnup ≤16,000 MWD/MTIHM
f. Initial Heavy Metal Weight:	≤ 27 kg/canister
g. Fuel Cladding O.D.:	≥ 0.412 inches
h. Fuel Cladding I.D.:	≤ 0.362 inches
i. Fuel Pellet O.D.:	≤ 0.358 inches
j. Active Fuel Length:	≤ 111 inches
k. Canister Weight:	≤ 550 lbs, including fuel

Table 2.1-1 (page 17 of ~~29~~30)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E and MPC-24EF

A. Allowable Contents

1. Uranium oxide, PWR INTACT FUEL ASSEMBLIES listed in Table 2.1-2, with or without NON-FUEL HARDWARE and meeting the following specifications (Note 1):

a. Cladding Type:	ZR or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class
b. Initial Enrichment:	As specified in Table 2.1-2 for the applicable fuel assembly array/class.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	Cooling time ≥ 8 years and an average burnup ≤ 40,000 MWD/MTU.
ii. All Other Array/Classes	Cooling time ≥ 3 years and an average burnup ≤ 68,200 MWD/MTU~~As specified in Section 2.4~~.
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

IV. MPC MODEL: MPC-24E and MPC-24EF (continued)

A. Allowable Contents (continued)

2. Uranium oxide, PWR DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS, with or without NON-FUEL HARDWARE, placed in DAMAGED FUEL CONTAINERS. Uranium oxide PWR DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS shall meet the criteria specified in Table 2.1-2 and meet the following specifications (Note 1):

a. Cladding Type:	ZR or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class
b. Initial Enrichment:	As specified in Table 2.1-2 for the applicable fuel assembly array/class.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	Cooling time ≥ 8 years and an average burnup ≤ 40,000 MWD/MTU.
ii. All Other Array/Classes	Cooling time ≥ 3 years and an average burnup ≤ 68,200 MWD/MTUAs specified in Section 2.4.
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 21 of ~~29~~30)
Fuel Assembly Limits

V. MPC MODEL: MPC-32 and MPC-32F

A. Allowable Contents

1. Uranium oxide, PWR INTACT FUEL ASSEMBLIES listed in Table 2.1-2, with or without NON-FUEL HARDWARE and meeting the following specifications (Note 1):

a. Cladding Type:	ZR or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class
b. Initial Enrichment:	As specified in Table 2.1-2 for the applicable fuel assembly array/class.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	Cooling time ≥ 9 years and an average burnup ≤ 30,000 MWD/MTU or cooling time ≥ 20 years and an average burnup ≤ 40,000 MWD/MTU.
ii. All Other Array/Classes	Cooling time ≥ 3 years and an average burnup ≤ 68,200 MWD/MTU~~As specified in Section 2.4~~.
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

V. MPC MODEL: MPC-32 and MPC-32F (cont'd)

A. Allowable Contents (cont'd)

2. Uranium oxide, PWR DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS, with or without NON-FUEL HARDWARE, placed in DAMAGED FUEL CONTAINERS. Uranium oxide PWR DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS shall meet the criteria specified in Table 2.1-2 and meet the following specifications (Note 1):

a. Cladding Type:	ZR or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class
b. Initial Enrichment:	As specified in Table 2.1-2 for the applicable fuel assembly array/class.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	Cooling time ≥ 9 years and an average burnup ≤ 30,000 MWD/MTU or cooling time ≥ 20 years and an average burnup ≤ 40,000 MWD/MTU.
ii. All Other Array/Classes	Cooling time ≥ 3 years and an average burnup ≤ 68,200 MWD/MTU~~As specified in Section 2.4~~.
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

VI. MPC MODEL: MPC-68M

A. Allowable Contents

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

a. Cladding Type:	ZR
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly	
i. Array/Class 8x8F	Cooling time ≥ 10 years and an average burnup ≤ 27,500 MWD/MTU.
ii. All Other Array/Classes	Cooling time ≥ 3 years and an average burnup ≤ 65,000 MWD/MTU~~As specified in Section 2.4~~.

Table 2.1-1 (page 27 of ~~29~~30)
Fuel Assembly Limits

VI. MPC MODEL: MPC-68M (continued)

A. Allowable Contents (continued)

2. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, with or without channels, placed in DAMAGED FUEL CONTAINERS. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS shall meet the criteria specified in Table 2.1-3, and meet the following specifications:

a. Cladding Type:	ZR
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Class 8x8F	Cooling time ≥ 10 years and an average burnup ≤ 27,500 MWD/MTU.
ii. All Other Array/Classes	Cooling time ≥ 3 years and an average burnup ≤ 65,000 MWD/MTU~~As specified in Section 2.4~~.

Table 2.1-1 (page 28 of ~~29~~30)
Fuel Assembly Limits

VI. MPC MODEL: MPC-68M (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly

i. Array/Class 8x8F	≤ 183.5 Watts
ii. All Other Array/Classes	As specified in Section 2.4.
f. Fuel Assembly Length	≤ 176.5 inches (nominal design)
g. Fuel Assembly Width	≤ 5.85 inches (nominal design)
h. Fuel Assembly Weight	≤ ~~7~~830 lbs, including channels and DFC

VI. MPC MODEL: MPC-68M (continued)

A. Allowable Contents (continued)

3. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:

a. Cladding type	ZR
b. Composition	98.2 wt.% ThO_2, 1.8 wt.% UO_2 with an enrichment of 93.5 wt.% ^{235}U OR 98.5 wt.% ThO_2, 1.5 wt.% UO_2 with an enrichment of 93.5% wt.% ^{235}U
c. Number of Rods per Thoria Rod Canister:	≤ 18
d. Decay Heat Per Thoria Rod Canister:	≤ 115 Watts
e. Post-irradiation Fuel Cooling Time and Average Burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time ≥ 18 years and an average burnup ≤16,000 MWD/MTIHM
f. Initial Heavy Metal Weight:	≤ 27 kg/canister
g. Fuel Cladding O.D.:	≥ 0.412 inches
h. Fuel Cladding I.D.:	≤ 0.362 inches
i. Fuel Pellet O.D.:	≤ 0.358 inches
j. Active Fuel Length:	≤ 111 inches
k. Canister Weight:	≤ 550 lbs, including fuel

Table 2.1-1 (page 30 of 30)
Fuel Assembly Limits

VI. MPC MODEL: MPC-68M (continued)

B. Quantity per MPC (up to a total of 68 assemblies)

1. Up to sixteen (16) DFCs containing BWR DAMAGED FUEL ASSEMBLIES and/or up to eight (8) DFCs containing FUEL DEBRIS. DFCs shall be located only in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68. The remaining fuel storage locations may be filled with Uranium Oxide BWR UNDAMAGED FUEL ASSEMBLIES.
2. Up to one (1) Dresden Unit 1 Thoria Rod Canister

Table 2.1-3 (page 4 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**10x10A**	**10x10B**	**10x10C**	**10x10D**	**10x10E**	**10x10F**	**10x10G**
Clad Material	ZR	ZR	ZR	SS	SS	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 188	≤ 188	≤ 179	≤ 125	≤ 125	≤ 192	≤ 188
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT(MPC-68, 68F, and 68FF) (wt.% ^{235}U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	Note 17	Note 17
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (MPC-68M) (wt.% ^{235}U) (Note 16, 19)	≤ 4.8	≤ 4.8	≤ 4.8	Note 18	Note 18	≤ 4.7 (Note 15)	≤ 4.~~6~~75 (Note 15)
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 8)	91/83 (Note 9)	96	100	96	92/78 (Note 8)	96/84
Fuel Rod Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940	≥ 0.4035	≥ 0.387
Fuel Rod Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500	≤ 0.3570	≤ 0.340
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430	≤ 0.3500	≤ 0.334
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557	≤ 0.510	≤ 0.512
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83	≤ 150	≤ 150
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4	2	5 (Note 10)
Water Rod Thickness (in.)	≥ 0.030	> 0.00	≥ 0.031	N/A	≥ 0.022	≥ 0.030	≥ 0.031
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080	≤ 0.120	≤ 0.060

Table 2.1-3 (page 5 of 5)
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. Deleted.
3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 1.5 percent for comparison with users' fuel records to account for manufacturer tolerances.
4. ≤ 0.635 wt. % 235U and ≤ 1.578 wt. % total fissile plutonium (239Pu and 241Pu), (wt. % of total fuel weight, i.e., UO2 plus PuO2).
5. This assembly class contains 74 total rods; 66 full length rods and 8 partial length rods.
6. Square, replacing nine fuel rods.
7. Variable.
8. This assembly contains 92 total fuel rods; 78 full length rods and 14 partial length rods.
9. This assembly class contains 91 total fuel rods; 83 full length rods and 8 partial length rods.
10. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.
11. These rods may also be sealed at both ends and contain Zr material in lieu of water.
12. This assembly is known as "QUAD+." It has four rectangular water cross segments dividing the assembly into four quadrants.
13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or the 9x9F set of limits for clad O.D., clad I.D., and pellet diameter.
14. For MPC-68, 68F, and 68FF loaded with both INTACT FUEL ASSEMBLIES and DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, the maximum PLANAR AVERAGE INITIAL ENRICHMENT for the INTACT FUEL ASSEMBLIES is limited to 3.7 wt.% ^{235}U, as applicable.
15. Fuel assemblies classified as damaged fuel assemblies are limited to 4.6 wt.% ^{235}U for the 10x10F and 10x10G arrays/classes and 4.0 wt.% ^{235}U for the 8x8F, 9x9E~~,~~ and 9x9F~~ and 10x10G~~ arrays/classes.
16. For MPC-68M loaded with both UNDAMAGED FUEL ASSEMBLIES and DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, the maximum PLANAR AVERAGE INITIAL ENRICHMENT for the UNDAMAGED FUEL ASSEMBLIES is limited to the enrichment limit of the damaged assembly.
17. This fuel assembly array/class is not allowable contents in MPC-68, 68F, or 68FF.
18. This fuel assembly array/class is not allowable contents in MPC-68M.
19. In accordance with the definition of UNDAMAGED FUEL ASSEMBLY, certain assemblies may be limited to up to 3.3 wt.% U-235. When loading these fuel

assemblies, all other undamaged fuel assemblies in the MPC are limited to ~~3.3 wt% U-235~~enrichments as specified in this table.

Table 2.1-8
NON-FUEL HARDWARE COOLING AND AVERAGE BURNUP (Notes 1, 2, 3, and 7)

Post-irradiation Cooling Time (years)	NSA with NFH INSERTS (Note 4) BURNUP (MWD/MTU)	NSA without NFH, GUIDE TUBE HARDWARE, or CONTROL COMPONENT (Note 5) BURNUP (MWD/MTU)	APSR BURNUP (MWD/MTU)
≥ 3	≤ 24,635	NA (Note 6)	NA
≥ 4	≤ 30,000	NA	NA
≥ 5	≤ 36,748	≤ 630,000	≤ 45,000
≥ 6	≤ 44,102	-	≤ 54,500
≥ 7	≤ 52,900	-	≤ 68,000
≥ 8	≤ 60,000	-	≤ 83,000
≥ 9	≤ 79,784-	-	≤ 111,000
≥ 10	≤ 101,826-	-	≤ 180,000
≥ 11	≤ 141,982-	-	≤ 630,000
≥ 12	≤ 360,000	-	-

Notes:
1. 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.
2. Linear interpolation between points is permitted, except that APSR burnups > 180,000 MWD/MTU and ≤ 630,000 MWD/MTU must be cooled ≥ 11 years.
3. Applicable to uniform loading and regionalized loading.
4. Includes Burnable Poison Rod Assemblies (BPRAs), Wet Annular Burnable Absorbers (WABAs), ~~and~~ vibration suppressor inserts and Neutron Source Assemblies (NSAs) in combination with other control components (i.e. BPRAs, TPDs, and/or RCCAs).
5. Includes Thimble Plug Devices (TPDs), water displacement guide tube plugs, orifice rod assemblies, Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), ~~and~~ Rod Cluster Control Assemblies (RCCAs)~~.~~ and NSAs without other forms of control components.
6. NA means not authorized for loading at this cooling time.
7. Non-fuel hardware burnup and cooling times are not applicable to ITTRs since they are installed post irradiation.

2.4 Decay Heat~~, Burnup, and Cooling Time~~ Limits for ZR-Clad Fuel

This section provides the limits on ZR-clad fuel assembly decay heat~~, burnup, and cooling time~~ for storage in the HI-STORM 100 System. Burnup and cooling time limits on ZR-clad fuel are provided in Table 2.1-1. ~~The method to calculate the limits and verify compliance, including examples, is provided in Chapter 12 of the HI-STORM 100 FSAR.~~

2.4.1 Uniform Fuel Loading Decay Heat Limits for ZR-clad fuel

Table 2.4-1 provides the maximum allowable decay heat per fuel storage location for ZR-clad fuel in uniform fuel loading for each MPC model.

Table 2.4-1

Maximum Allowable Decay Heat per Fuel Storage Location

(Uniform Loading, ZR-Clad)

MPC Model	**Decay Heat per Fuel Storage Location (kW)**	
	Intact or Undamaged Fuel Assemblies	Damaged Fuel Assemblies and Fuel Debris
MPC-24	$\leq$ 1.416	Not Permitted
MPC-24E/24EF	$\leq$ 1.416	$\leq$ 1.114
MPC-32/32F	$\leq$ 1.062	$\leq$ 0.718
MPC-68/68FF/68M	$\leq$ 0.500	$\leq$ 0.393

2.4.3 DeletedBurnup Limits as a Function of Cooling Time for ZR-Clad Fuel

The maximum allowable fuel assembly average burnup varies with the following parameters:

- Minimum fuel assembly cooling time
- Maximum fuel assembly decay heat
- Minimum fuel assembly average enrichment

The maximum allowable ZR-clad fuel assembly average burnup for a given MINIMUM ENRICHMENT is calculated as described below for minimum cooling times between 3 and 20 years using the maximum permissible decay heat determined in Section 2.4.1 or 2.4.2. Different fuel assembly average burnup limits may be calculated for different minimum enrichments (by individual fuel assembly) for use in choosing the fuel assemblies to be loaded into a given MPC.

2.4.3.1 Choose a fuel assembly minimum enrichment, E_{235}.

2.4.3.2 Calculate the maximum allowable fuel assembly average burnup for a minimum cooling time between 3 and 20 years using the equation below.

$$Bu = (A \times q) + (B \times q^2) + (C \times q^3) + [D \times (E_{235})^2] + (E \times q \times E_{235}) + (F \times q^2 \times E_{235}) + G$$

Where:

Bu = Maximum allowable average burnup per fuel assembly (MWD/MTU)

q = Maximum allowable decay heat per fuel storage location determined in Section 2.4.1 or 2.4.2 (kW)

E_{235} = Minimum fuel assembly average enrichment (wt. % ^{235}U) (e.g., for 4.05 wt.%, use 4.05)

A through G = Coefficients from Tables 2.4-3 and 2.4-4 for the applicable fuel assembly array/class and minimum cooling time

2.4.3.3 Calculated burnup limits shall be rounded down to the nearest integer.

2.4.3.4 Calculated burnup limits greater than 68,200 MWD/MTU for PWR fuel and 65,000 MWD/MTU for BWR must be reduced to be equal to these values.

2.4.3.5 Linear interpolation of calculated burnups between cooling times for a given fuel assembly maximum decay heat and minimum enrichment is permitted. For example, the allowable burnup for a cooling time of 4.5 years may be interpolated between those burnups calculated for 4 year and 5 years.

2.4.3.6 ~~Each ZR clad fuel assembly to be stored must have a MINIMUM ENRICHMENT greater than or equal to the value used in Step 2.4.3.2.~~

2.4.4 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.

Table 2.4-3 (Page 1 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 14x14A						
	A	B	C	D	E	F	G
≥ 3	19311.5	275.367	-59.0252	-139.41	2851.12	-451.845	-615.413
≥ 4	33865.9	-5473.03	851.121	-132.739	3408.58	-656.479	-609.523
≥ 5	46686.2	-13226.9	2588.39	-150.149	3871.87	-806.533	-90.2065
≥ 6	56328.9	-20443.2	4547.38	-176.815	4299.19	-927.358	603.192
≥ 7	64136	-27137.5	6628.18	-200.933	4669.22	-1018.94	797.162
≥ 8	71744.1	-34290.3	9036.9	-214.249	4886.95	-1037.59	508.703
≥ 9	77262	-39724.2	11061	-228.2	5141.35	-1102.05	338.294
≥ 10	82939.8	-45575.6	13320.2	-233.691	5266.25	-1095.94	-73.3159
≥ 11	86541	-49289.6	14921.7	-242.092	5444.54	-1141.6	-83.0603
≥ 12	91383	-54456.7	17107	-242.881	5528.7	-1149.2	-547.579
≥ 13	95877.6	-59404.7	19268	-240.36	5524.35	-1094.72	-933.64
≥ 14	97648.3	-61091.6	20261.7	-244.234	5654.56	-1151.47	-749.836
≥ 15	102533	-66651.5	22799.7	-240.858	5647.05	-1120.32	-1293.34
≥ 16	106216	-70753.8	24830.1	-237.04	5647.63	-1099.12	-1583.89
≥ 17	109863	-75005	27038	-234.299	5652.45	-1080.98	-1862.07
≥ 18	111460	-76482.3	28076.5	-234.426	5703.52	-1104.39	-1695.77
≥ 19	114916	-80339.6	30126.5	-229.73	5663.21	-1065.48	-1941.83
≥ 20	119592	-86161.5	33258.2	-227.256	5700.49	-1100.21	-2474.01

Table 2.4-3 (Page 2 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 14x14B						
	A	B	C	D	E	F	G
≥ 3	18036.1	63.7639	-24.7251	-130.732	2449.87	-347.748	-858.192
≥ 4	30303.4	-4304.2	598.79	-118.757	2853.18	-486.453	-459.902
≥ 5	40779.6	-9922.93	1722.83	-138.174	3255.69	-608.267	245.251
≥ 6	48806.7	-15248.9	3021.47	-158.69	3570.24	-689.876	833.917
≥ 7	55070.5	-19934.6	4325.62	-179.964	3870.33	-765.849	1203.89
≥ 8	60619.6	-24346	5649.29	-189.701	4042.23	-795.324	1158.12
≥ 9	64605.7	-27677.1	6778.12	-205.459	4292.35	-877.966	1169.88
≥ 10	69083.8	-31509.4	8072.42	-206.157	4358.01	-875.041	856.449
≥ 11	72663.2	-34663.9	9228.96	-209.199	4442.68	-889.512	671.567
≥ 12	74808.9	-36367	9948.88	-214.344	4571.29	-942.418	765.261
≥ 13	78340.3	-39541.1	11173.8	-212.8	4615.06	-957.833	410.807
≥ 14	81274.8	-42172.3	12259.9	-209.758	4626.13	-958.016	190.59
≥ 15	83961.4	-44624.5	13329.1	-207.697	4632.16	-952.876	20.8575
≥ 16	84968.5	-44982.1	13615.8	-207.171	4683.41	-992.162	247.54
≥ 17	87721.6	-47543.1	14781.4	-203.373	4674.3	-988.577	37.9689
≥ 18	90562.9	-50100.4	15940.4	-198.649	4651.64	-982.459	-247.421
≥ 19	93011.6	-52316.6	17049.9	-194.964	4644.76	-994.63	-413.021
≥ 20	95567.8	-54566.6	18124	-190.22	4593.92	-963.412	-551.983

Table 2.4-3 (Page 3 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 14x14C						
	A	B	C	D	E	F	G
≥ 3	18263.7	174.161	-57.6694	-138.112	2539.74	-369.764	-1372.33
≥ 4	30514.5	-4291.52	562.37	-124.944	2869.17	-481.139	-889.883
≥ 5	41338	-10325.7	1752.96	-141.247	3146.48	-535.709	-248.078
≥ 6	48969.7	-15421.3	2966.33	-163.574	3429.74	-587.225	429.331
≥ 7	55384.6	-20228.9	4261.47	-180.846	3654.55	-617.255	599.251
≥ 8	60240.2	-24093.2	5418.86	-199.974	3893.72	-663.995	693.934
≥ 9	64729	-27745.7	6545.45	-205.385	3986.06	-650.124	512.528
≥ 10	68413.7	-30942.2	7651.29	-216.408	4174.71	-702.931	380.431
≥ 11	71870.6	-33906.7	8692.81	-218.813	4248.28	-704.458	160.645
≥ 12	74918.4	-36522	9660.01	-218.248	4283.68	-696.498	-29.0682
≥ 13	77348.3	-38613.7	10501.8	-220.644	4348.23	-702.266	-118.646
≥ 14	79817.1	-40661.8	11331.2	-218.711	4382.32	-710.578	-236.123
≥ 15	82354.2	-42858.3	12257.3	-215.835	4405.89	-718.805	-431.051
≥ 16	84787.2	-44994.5	13185.9	-213.386	4410.99	-711.437	-572.104
≥ 17	87084.6	-46866.1	14004.8	-206.788	4360.3	-679.542	-724.721
≥ 18	88083.1	-47387.1	14393.4	-208.681	4420.85	-709.311	-534.454
≥ 19	90783.6	-49760.6	15462.7	-203.649	4403.3	-705.741	-773.066
≥ 20	93212	-51753.3	16401.5	-197.232	4361.65	-692.925	-964.628

Table 2.4-3 (Page 4 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 15x15A/B/C						
	A	B	C	D	E	F	G
≥ 3	15037.3	108.689	-18.8378	-127.422	2050.02	-242.828	-580.66
≥ 4	25506.6	-2994.03	356.834	-116.45	2430.25	-350.901	-356.378
≥ 5	34788.8	-7173.07	1065.9	-124.785	2712.23	-424.681	267.705
≥ 6	41948.6	-11225.3	1912.12	-145.727	3003.29	-489.538	852.112
≥ 7	47524.9	-14770.9	2755.16	-165.889	3253.9	-542.7	1146.96
≥ 8	52596.9	-18348.8	3699.72	-177.17	3415.69	-567.012	1021.41
≥ 9	56055.4	-20837.1	4430.93	-192.168	3625.93	-623.325	1058.61
≥ 10	59611.3	-23402.1	5179.52	-195.105	3699.18	-626.448	868.517
≥ 11	62765.3	-25766.5	5924.71	-195.57	3749.91	-627.139	667.124
≥ 12	65664.4	-28004.8	6670.75	-195.08	3788.33	-628.904	410.783
≥ 13	67281.7	-29116.7	7120.59	-202.817	3929.38	-688.738	492.309
≥ 14	69961.4	-31158.6	7834.02	-197.988	3917.29	-677.565	266.561
≥ 15	72146	-32795.7	8453.67	-195.083	3931.47	-681.037	99.0606
≥ 16	74142.6	-34244.8	9023.57	-190.645	3905.54	-663.682	10.8885
≥ 17	76411.4	-36026.3	9729.98	-188.874	3911.21	-663.449	-151.805
≥ 18	77091	-36088	9884.09	-188.554	3965.08	-708.55	59.3839
≥ 19	79194.5	-37566.4	10477.5	-181.656	3906.93	-682.4	-117.952
≥ 20	81600.4	-39464.5	11281.9	-175.182	3869.49	-677.179	-367.705

Table 2.4-3 (Page 5 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 15x15D/E/F/H/I						
	A	B	C	D	E	F	G
≥3	14376.7	102.205	-20.6279	-126.017	1903.36	-210.883	-493.065
≥4	24351.4	-2686.57	297.975	-110.819	2233.78	-301.615	-152.713
≥5	33518.4	-6711.35	958.544	-122.85	2522.7	-371.286	392.608
≥6	40377	-10472.4	1718.53	-144.535	2793.29	-426.436	951.528
≥7	46105.8	-13996.2	2515.32	-157.827	2962.46	-445.314	1100.56
≥8	50219.7	-16677.7	3198.3	-175.057	3176.74	-492.727	1223.62
≥9	54281.2	-19555.6	3983.47	-181.703	3279.03	-499.997	1034.55
≥10	56761.6	-21287.3	4525.98	-195.045	3470.41	-559.074	1103.3
≥11	59820	-23445.2	5165.43	-194.997	3518.23	-561.422	862.68
≥12	62287.2	-25164.6	5709.9	-194.771	3552.69	-561.466	680.488
≥13	64799	-27023.7	6335.16	-192.121	3570.41	-561.326	469.583
≥14	66938.7	-28593.1	6892.63	-194.226	3632.92	-583.997	319.867
≥15	68116.5	-29148.6	7140.09	-192.545	3670.39	-607.278	395.344
≥16	70154.9	-30570.1	7662.91	-187.366	3649.14	-597.205	232.318
≥17	72042.5	-31867.6	8169.01	-183.453	3646.92	-603.907	96.0388
≥18	73719.8	-32926.1	8596.12	-177.896	3614.57	-592.868	46.6774
≥19	75183.1	-33727.4	8949.64	-172.386	3581.13	-586.347	3.57256
≥20	77306.1	-35449	9690.02	-173.784	3636.87	-626.321	-205.513

Table 2.4-3 (Page 6 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 16X16A						
	A	B	C	D	E	F	G
≥ 3	16226.8	143.714	-32.4809	-136.707	2255.33	-291.683	-699.947
≥ 4	27844.2	-3590.69	444.838	-124.301	2644.09	-411.598	-381.106
≥ 5	38191.5	-8678.48	1361.58	-132.855	2910.45	-473.183	224.473
≥ 6	46382.2	-13819.6	2511.32	-158.262	3216.92	-532.337	706.656
≥ 7	52692.3	-18289	3657.18	-179.765	3488.3	-583.133	908.839
≥ 8	57758.7	-22133.7	4736.88	-199.014	3717.42	-618.83	944.903
≥ 9	62363.3	-25798.7	5841.18	-207.025	3844.38	-625.741	734.928
≥ 10	66659.1	-29416.3	6993.31	-216.458	3981.97	-642.641	389.366
≥ 11	69262.7	-31452.7	7724.66	-220.836	4107.55	-681.043	407.121
≥ 12	72631.5	-34291.9	8704.8	-219.929	4131.5	-662.513	100.093
≥ 13	75375.3	-36589.3	9555.88	-217.994	4143.15	-644.014	-62.3294
≥ 14	78178.7	-39097.1	10532	-221.923	4226.28	-667.012	-317.743
≥ 15	79706.3	-40104	10993.3	-218.751	4242.12	-670.665	-205.579
≥ 16	82392.6	-42418.9	11940.7	-216.278	4274.09	-689.236	-479.752
≥ 17	84521.8	-44150.5	12683.3	-212.056	4245.99	-665.418	-558.901
≥ 18	86777.1	-45984.8	13479	-204.867	4180.8	-621.805	-716.366
≥ 19	89179.7	-48109.8	14434.5	-206.484	4230.03	-648.557	-902.1
≥ 20	90141.7	-48401.4	14702.6	-203.284	4245.54	-670.655	-734.604

Table 2.4-3 (Page 7 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 17x17A/16x16B/C						
	A	B	C	D	E	F	G
≥3	15985.1	3.53963	-9.04955	-128.835	2149.5	-260.415	-262.997
≥4	27532.9	-3494.41	428.199	-119.504	2603.01	-390.91	-140.319
≥5	38481.2	-8870.98	1411.03	-139.279	3008.46	-492.881	388.377
≥6	47410.9	-14479.6	2679.08	-162.13	3335.48	-557.777	702.164
≥7	54596.8	-19703.2	4043.46	-181.339	3586.06	-587.634	804.05
≥8	60146.1	-24003.4	5271.54	-201.262	3830.32	-621.706	848.454
≥9	65006.3	-27951	6479.04	-210.753	3977.69	-627.805	615.84
≥10	69216	-31614.7	7712.58	-222.423	4173.4	-672.33	387.879
≥11	73001.3	-34871.1	8824.44	-225.128	4238.28	-657.259	101.654
≥12	76326.1	-37795.9	9887.35	-226.731	4298.11	-647.55	-122.236
≥13	78859.9	-40058.9	10797.1	-231.798	4402.14	-669.982	-203.383
≥14	82201.3	-43032.5	11934.1	-228.162	4417.99	-661.61	-561.969
≥15	84950	-45544.6	12972.4	-225.369	4417.84	-637.422	-771.254
≥16	87511.8	-47720	13857.7	-219.255	4365.24	-585.655	-907.775
≥17	90496.4	-50728.9	15186	-223.019	4446.51	-613.378	-1200.94
≥18	91392.5	-51002.4	15461.4	-220.272	4475.28	-636.398	-1003.81
≥19	94343.9	-53670.8	16631.6	-214.045	4441.31	-616.201	-1310.01
≥20	96562.9	-55591.2	17553.4	-209.917	4397.67	-573.199	-1380.64

Table 2.4-3 (Page 8 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 17x17B/C						
	A	B	C	D	E	F	G
≥3	14738	47.5402	-13.8187	-127.895	1946.58	-219.289	-389.029
≥4	25285.2	-3011.92	350.116	-115.75	2316.89	-319.23	-220.413
≥5	34589.6	-7130.34	1037.26	-128.673	2627.27	-394.58	459.642
≥6	42056.2	-11353.7	1908.68	-150.234	2897.38	-444.316	923.971
≥7	47977.6	-15204.8	2827.4	-173.349	3178.25	-504.16	1138.82
≥8	52924	-18547.6	3671.08	-183.025	3298.64	-501.278	1064.68
≥9	56465.5	-21139.4	4435.67	-200.386	3538	-569.712	1078.78
≥10	60190.9	-23872.7	5224.31	-203.233	3602.88	-562.312	805.336
≥11	63482.1	-26431.1	6035.79	-205.096	3668.84	-566.889	536.011
≥12	66095	-28311.8	6637.72	-204.367	3692.68	-555.305	372.223
≥13	67757.4	-29474.4	7094.08	-211.649	3826.42	-606.886	437.412
≥14	70403.7	-31517.4	7807.15	-207.668	3828.69	-601.081	183.09
≥15	72506.5	-33036.1	8372.59	-203.428	3823.38	-594.995	47.5175
≥16	74625.2	-34620.5	8974.32	-199.003	3798.57	-573.098	-95.0221
≥17	76549	-35952.6	9498.14	-193.459	3766.52	-556.928	-190.662
≥18	77871.9	-36785.5	9916.91	-195.592	3837.65	-599.45	-152.261
≥19	79834.8	-38191.6	10501.9	-190.83	3812.46	-589.635	-286.847
≥20	81975.5	-39777.2	11174.5	-185.767	3795.78	-595.664	-475.978

Table 2.4-4 (Page 1 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 7x7B & 10x10F						
	A	B	C	D	E	F	G
≥3	26409.1	28347.5	-16858	-147.076	5636.32	-1606.75	1177.88
≥4	61967.8	-6618.31	-4131.96	-113.949	6122.77	-2042.85	-96.7439
≥5	91601.1	-49298.3	17826.5	-132.045	6823.14	-2418.49	-185.189
≥6	111369	-80890.1	35713.8	-150.262	7288.51	-2471.1	86.6363
≥7	126904	-108669	53338.1	-167.764	7650.57	-2340.78	150.403
≥8	139181	-132294	69852.5	-187.317	8098.66	-2336.13	97.5285
≥9	150334	-154490	86148.1	-193.899	8232.84	-2040.37	-123.029
≥10	159897	-173614	100819	-194.156	8254.99	-1708.32	-373.605
≥11	166931	-186860	111502	-193.776	8251.55	-1393.91	-543.677
≥12	173691	-201687	125166	-202.578	8626.84	-1642.3	-650.814
≥13	180312	-215406	137518	-201.041	8642.19	-1469.45	-810.024
≥14	185927	-227005	148721	-197.938	8607.6	-1225.95	-892.876
≥15	191151	-236120	156781	-191.625	8451.86	-846.27	-1019.4
≥16	195761	-244598	165372	-187.043	8359.19	-572.561	-1068.19
≥17	200791	-256573	179816	-197.26	8914.28	-1393.37	-1218.63
≥18	206068	-266136	188841	-187.191	8569.56	-730.898	-1363.79
≥19	210187	-273609	197794	-182.151	8488.23	-584.727	-1335.59
≥20	213731	-278120	203074	-175.864	8395.63	-457.304	-1364.38

Table 2.4-4 (Page 2 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 8x8B						
	A	B	C	D	E	F	G
≥ 3	28219.6	28963.7	-17616.2	-147.68	5887.41	-1730.96	1048.21
≥ 4	66061.8	-10742.4	-1961.82	-123.066	6565.54	-2356.05	-298.005
≥ 5	95790.7	-53401.7	19836.7	-134.584	7145.41	-2637.09	-298.858
≥ 6	117477	-90055.9	41383.9	-154.758	7613.43	-2612.69	-64.9921
≥ 7	134090	-120643	60983	-168.675	7809	-2183.3	-40.8885
≥ 8	148186	-149181	81418.7	-185.726	8190.07	-2040.31	-260.773
≥ 9	159082	-172081	99175.2	-197.185	8450.86	-1792.04	-381.705
≥ 10	168816	-191389	113810	-195.613	8359.87	-1244.22	-613.594
≥ 11	177221	-210599	131099	-208.3	8810	-1466.49	-819.773
≥ 12	183929	-224384	143405	-207.497	8841.33	-1227.71	-929.708
≥ 13	191093	-240384	158327	-204.95	8760.17	-811.708	-1154.76
≥ 14	196787	-252211	169664	-204.574	8810.95	-610.928	-1208.97
≥ 15	203345	-267656	186057	-208.962	9078.41	-828.954	-1383.76
≥ 16	207973	-276838	196071	-204.592	9024.17	-640.808	-1436.43
≥ 17	213891	-290411	211145	-202.169	9024.19	-482.1	-1595.28
≥ 18	217483	-294066	214600	-194.243	8859.35	-244.684	-1529.61
≥ 19	220504	-297897	219704	-190.161	8794.97	-10.9863	-1433.86
≥ 20	227821	-318395	245322	-194.682	9060.96	-350.308	-1741.16

Table 2.4-4 (Page 3 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 8x8C/D/E						
	A	B	C	D	E	F	G
≥ 3	28592.7	28691.5	-17773.6	-149.418	5969.45	-1746.07	1063.62
≥ 4	66720.8	-12115.7	-1154	-128.444	6787.16	-2529.99	-302.155
≥ 5	96929.1	-55827.5	21140.3	-136.228	7259.19	-2685.06	-334.328
≥ 6	118190	-92000.2	42602.5	-162.204	7907.46	-2853.42	-47.5465
≥ 7	135120	-123437	62827.1	-172.397	8059.72	-2385.81	-75.0053
≥ 8	149162	-152986	84543.1	-195.458	8559.11	-2306.54	-183.595
≥ 9	161041	-177511	103020	-200.087	8632.84	-1864.4	-433.081
≥ 10	171754	-201468	122929	-209.799	8952.06	-1802.86	-755.742
≥ 11	179364	-217723	137000	-215.803	9142.37	-1664.82	-847.268
≥ 12	186090	-232150	150255	-216.033	9218.36	-1441.92	-975.817
≥ 13	193571	-249160	165997	-213.204	9146.99	-1011.13	-1119.47
≥ 14	200034	-263671	180359	-210.559	9107.54	-694.626	-1312.55
≥ 15	205581	-275904	193585	-216.242	9446.57	-1040.65	-1428.13
≥ 16	212015	-290101	207594	-210.036	9212.93	-428.321	-1590.7
≥ 17	216775	-299399	218278	-204.611	9187.86	-398.353	-1657.6
≥ 18	220653	-306719	227133	-202.498	9186.34	-181.672	-1611.86
≥ 19	224859	-314004	235956	-193.902	8990.14	145.151	-1604.71
≥ 20	228541	-320787	245449	-200.727	9310.87	-230.252	-1570.18

Table 2.4-4 (Page 4 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 9x9A						
	A	B	C	D	E	F	G
≥ 3	30538.7	28463.2	-18105.5	-150.039	6226.92	-1876.69	1034.06
≥ 4	71040.1	-16692.2	1164.15	-128.241	7105.27	-2728.58	-414.09
≥ 5	100888	-60277.7	24150.1	-142.541	7896.11	-3272.86	-232.197
≥ 6	124846	-102954	50350.8	-161.849	8350.16	-3163.44	-91.1396
≥ 7	143516	-140615	76456.5	-185.538	8833.04	-2949.38	-104.802
≥ 8	158218	-171718	99788.2	-196.315	9048.88	-2529.26	-259.929
≥ 9	172226	-204312	126620	-214.214	9511.56	-2459.19	-624.954
≥ 10	182700	-227938	146736	-215.793	9555.41	-1959.92	-830.943
≥ 11	190734	-246174	163557	-218.071	9649.43	-1647.5	-935.021
≥ 12	199997	-269577	186406	-223.975	9884.92	-1534.34	-1235.27
≥ 13	207414	-287446	204723	-228.808	10131.7	-1614.49	-1358.61
≥ 14	215263	-306131	223440	-220.919	9928.27	-988.276	-1638.05
≥ 15	221920	-321612	239503	-217.949	9839.02	-554.709	-1784.04
≥ 16	226532	-331778	252234	-216.189	9893.43	-442.149	-1754.72
≥ 17	232959	-348593	272609	-219.907	10126.3	-663.84	-1915.3
≥ 18	240810	-369085	296809	-219.729	10294.6	-859.302	-2218.87
≥ 19	244637	-375057	304456	-210.997	10077.8	-425.446	-2127.83
≥ 20	248112	-379262	309391	-204.191	9863.67	100.27	-2059.39

Table 2.4-4 (Page 5 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 9x9B						
	A	B	C	D	E	F	G
≥ 3	30613.2	28985.3	-18371	-151.117	6321.55	-1881.28	988.92
≥ 4	71346.6	-15922.9	631.132	-128.876	7232.47	-2810.64	-471.737
≥ 5	102131	-60654.1	23762.7	-140.748	7881.6	-3156.38	-417.979
≥ 6	127187	-105842	51525.2	-162.228	8307.4	-2913.08	-342.13
≥ 7	146853	-145834	79146.5	-185.192	8718.74	-2529.57	-484.885
≥ 8	162013	-178244	103205	-197.825	8896.39	-1921.58	-584.013
≥ 9	176764	-212856	131577	-215.41	9328.18	-1737.12	-1041.11
≥ 10	186900	-235819	151238	-218.98	9388.08	-1179.87	-1202.83
≥ 11	196178	-257688	171031	-220.323	9408.47	-638.53	-1385.16
≥ 12	205366	-280266	192775	-223.715	9592.12	-472.261	-1661.6
≥ 13	215012	-306103	218866	-231.821	9853.37	-361.449	-1985.56
≥ 14	222368	-324558	238655	-228.062	9834.57	3.47358	-2178.84
≥ 15	226705	-332738	247316	-224.659	9696.59	632.172	-2090.75
≥ 16	233846	-349835	265676	-221.533	9649.93	913.747	-2243.34
≥ 17	243979	-379622	300077	-222.351	9792.17	1011.04	-2753.36
≥ 18	247774	-386203	308873	-220.306	9791.37	1164.58	-2612.25
≥ 19	254041	-401906	327901	-213.96	9645.47	1664.94	-2786.2
≥ 20	256003	-402034	330566	-215.242	9850.42	1359.46	-2550.06

Table 2.4-4 (Page 6 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 9x9C/D						
	A	B	C	D	E	F	G
≥ 3	30051.6	29548.7	-18614.2	-148.276	6148.44	-1810.34	1006
≥ 4	70472.7	-14696.6	-233.567	-127.728	7008.69	-2634.22	-444.373
≥ 5	101298	-59638.9	23065.2	-138.523	7627.57	-2958.03	-377.965
≥ 6	125546	-102740	49217.4	-160.811	8096.34	-2798.88	-259.767
≥ 7	143887	-139261	74100.4	-184.302	8550.86	-2517.19	-275.151
≥ 8	159633	-172741	98641.4	-194.351	8636.89	-1838.81	-486.731
≥ 9	173517	-204709	124803	-212.604	9151.98	-1853.27	-887.137
≥ 10	182895	-225481	142362	-218.251	9262.59	-1408.25	-978.356
≥ 11	192530	-247839	162173	-217.381	9213.58	-818.676	-1222.12
≥ 12	201127	-268201	181030	-215.552	9147.44	-232.221	-1481.55
≥ 13	209538	-289761	203291	-225.092	9588.12	-574.227	-1749.35
≥ 14	216798	-306958	220468	-222.578	9518.22	-69.9307	-1919.71
≥ 15	223515	-323254	237933	-217.398	9366.52	475.506	-2012.93
≥ 16	228796	-334529	250541	-215.004	9369.33	662.325	-2122.75
≥ 17	237256	-356311	273419	-206.483	9029.55	1551.3	-2367.96
≥ 18	242778	-369493	290354	-215.557	9600.71	659.297	-2589.32
≥ 19	246704	-377971	302630	-210.768	9509.41	1025.34	-2476.06
≥ 20	249944	-382059	308281	-205.495	9362.63	1389.71	-2350.49

Table 2.4-4 (Page 7 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 9x9E/F						
	A	B	C	D	E	F	G
≥3	30284.3	26949.5	-16926.4	-147.914	6017.02	-1854.81	1026.15
≥4	69727.4	-17117.2	1982.33	-127.983	6874.68	-2673.01	-359.962
≥5	98438.9	-58492	23382.2	-138.712	7513.55	-3038.23	-112.641
≥6	119765	-95024.1	45261	-159.669	8074.25	-3129.49	221.182
≥7	136740	-128219	67940.1	-182.439	8595.68	-3098.17	315.544
≥8	150745	-156607	88691.5	-193.941	8908.73	-2947.64	142.072
≥9	162915	-182667	109134	-198.37	8999.11	-2531	-93.4908
≥10	174000	-208668	131543	-210.777	9365.52	-2511.74	-445.876
≥11	181524	-224252	145280	-212.407	9489.67	-2387.49	-544.123
≥12	188946	-240952	160787	-210.65	9478.1	-2029.94	-652.339
≥13	193762	-250900	171363	-215.798	9742.31	-2179.24	-608.636
≥14	203288	-275191	196115	-218.113	9992.5	-2437.71	-1065.92
≥15	208108	-284395	205221	-213.956	9857.25	-1970.65	-1082.94
≥16	215093	-301828	224757	-209.736	9789.58	-1718.37	-1303.35
≥17	220056	-310906	234180	-201.494	9541.73	-1230.42	-1284.15
≥18	224545	-320969	247724	-206.807	9892.97	-1790.61	-1381.9
≥19	226901	-322168	250395	-204.073	9902.14	-1748.78	-1253.22
≥20	235561	-345414	276856	-198.306	9720.78	-1284.14	-1569.18

Table 2.4-4 (Page 8 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 9x9G						
	A	B	C	D	E	F	G
≥ 3	35158.5	26918.5	-17976.7	-149.915	6787.19	-2154.29	836.894
≥ 4	77137.2	-19760.1	2371.28	-130.934	8015.43	-3512.38	-455.424
≥ 5	113405	-77931.2	35511.2	-150.637	8932.55	-4099.48	-629.806
≥ 6	139938	-128700	68698.3	-173.799	9451.22	-3847.83	-455.905
≥ 7	164267	-183309	109526	-193.952	9737.91	-3046.84	-737.992
≥ 8	182646	-227630	146275	-210.936	10092.3	-2489.3	-1066.96
≥ 9	199309	-270496	184230	-218.617	10124.3	-1453.81	-1381.41
≥ 10	213186	-308612	221699	-235.828	10703.2	-1483.31	-1821.73
≥ 11	225587	-342892	256242	-236.112	10658.5	-612.076	-2134.65
≥ 12	235725	-370471	285195	-234.378	10604.9	118.591	-2417.89
≥ 13	247043	-404028	323049	-245.79	11158.2	-281.813	-2869.82
≥ 14	253649	-421134	342682	-243.142	11082.3	400.019	-2903.88
≥ 15	262750	-448593	376340	-245.435	11241.2	581.355	-3125.07
≥ 16	270816	-470846	402249	-236.294	10845.4	1791.46	-3293.07
≥ 17	279840	-500272	441964	-241.324	11222.6	1455.84	-3528.25
≥ 18	284533	-511287	458538	-240.905	11367.2	1459.68	-3520.94
≥ 19	295787	-545885	501824	-235.685	11188.2	2082.21	-3954.2
≥ 20	300209	-556936	519174	-229.539	10956	2942.09	-3872.87

Table 2.4-4 (Page 9 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 10x10A/B/G						
	A	B	C	D	E	F	G
≥ 3	29285.4	27562.2	-16985	-148.415	5960.56	-1810.79	1001.45
≥ 4	67844.9	-14383	395.619	-127.723	6754.56	-2547.96	-369.267
≥ 5	96660.5	-55383.8	21180.4	-137.17	7296.6	-2793.58	-192.85
≥ 6	118098	-91995	42958	-162.985	7931.44	-2940.84	60.9197
≥ 7	135115	-123721	63588.9	-171.747	8060.23	-2485.59	73.6219
≥ 8	148721	-151690	84143.9	-190.26	8515.81	-2444.25	-63.4649
≥ 9	160770	-177397	104069	-197.534	8673.6	-2101.25	-331.046
≥ 10	170331	-198419	121817	-213.692	9178.33	-2351.54	-472.844
≥ 11	179130	-217799	138652	-209.75	9095.43	-1842.88	-705.254
≥ 12	186070	-232389	151792	-208.946	9104.52	-1565.11	-822.73
≥ 13	192407	-246005	164928	-209.696	9234.7	-1541.54	-979.245
≥ 14	200493	-265596	183851	-207.639	9159.83	-1095.72	-1240.61
≥ 15	205594	-276161	195760	-213.491	9564.23	-1672.22	-1333.64
≥ 16	209386	-282942	204110	-209.322	9515.83	-1506.86	-1286.82
≥ 17	214972	-295149	217095	-202.445	9292.34	-893.6	-1364.97
≥ 18	219312	-302748	225826	-198.667	9272.27	-878.536	-1379.58
≥ 19	223481	-310663	235908	-194.825	9252.9	-785.066	-1379.62
≥ 20	227628	-319115	247597	-199.194	9509.02	-1135.23	-1386.19

Table 2.4-4 (Page 10 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 10x10C						
	A	B	C	D	E	F	G
≥ 3	31425.3	27358.9	-17413.3	-152.096	6367.53	-1967.91	925.763
≥ 4	71804	-16964.1	1000.4	-129.299	7227.18	-2806.44	-416.92
≥ 5	102685	-62383.3	24971.2	-142.316	7961	-3290.98	-354.784
≥ 6	126962	-105802	51444.6	-164.283	8421.44	-3104.21	-186.615
≥ 7	146284	-145608	79275.5	-188.967	8927.23	-2859.08	-251.163
≥ 8	162748	-181259	105859	-199.122	9052.91	-2206.31	-554.124
≥ 9	176612	-214183	133261	-217.56	9492.17	-1999.28	-860.669
≥ 10	187756	-239944	155315	-219.56	9532.45	-1470.9	-1113.42
≥ 11	196580	-260941	174536	-222.457	9591.64	-944.473	-1225.79
≥ 12	208017	-291492	204805	-233.488	10058.3	-1217.01	-1749.84
≥ 13	214920	-307772	221158	-234.747	10137.1	-897.23	-1868.04
≥ 14	222562	-326471	240234	-228.569	9929.34	-183.47	-2016.12
≥ 15	228844	-342382	258347	-226.944	9936.76	117.061	-2106.05
≥ 16	233907	-353008	270390	-223.179	9910.72	360.39	-2105.23
≥ 17	244153	-383017	304819	-227.266	10103.2	380.393	-2633.23
≥ 18	249240	-395456	321452	-226.989	10284.1	169.947	-2623.67
≥ 19	254343	-406555	335240	-220.569	10070.5	764.689	-2640.2
≥ 20	260202	-421069	354249	-216.255	10069.9	854.497	-2732.77

DESIGN FEATURES

3.3.1 Alternatives to Codes, Standards, and Criteria

Table 3-1 lists approved alternatives to the ASME Code for the design of the MPCs, OVERPACKs, and TRANSFER CASKs of the HI-STORM 100 Cask System.

3.3.2 Construction/Fabrication Alternatives to Codes, Standards, and Criteria

Proposed alternatives to the ASME Code, Sections II and III, 1995 Edition with Addenda through 1997 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, 1995 Edition with Addenda through 1997, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

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

(continued)

3.4 Site-Specific Parameters and Analyses (continued)

NOTE: The above anchorage specifications are required for the seismic spectra defined in item 3.4.3.c.i. Users may use fewer studs or those of different diameter to account for site-specific seismic spectra less severe than those specified above. The embedment design shall comply with Appendix B of ACI-349-97. A later edition of this Code may be used, provided a written reconciliation is performed.

iii. Embedment Concrete Compressive Strength: ≥ 4,000 psi at 28 days

4. The analyzed flood condition of 15 fps water velocity and a height of 125 feet of water (full submergence of the loaded cask) are not exceeded.

5. 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.

6. a. For freestanding casks, the ISFSI pad shall be verified by analysis to limit cask deceleration during design basis drop and non-mechanistic tip-over events to ≤ 45 g's at the top of the MPC fuel basket. Analyses shall be performed using methodologies consistent with those described in the HI-STORM 100 FSAR. A restriction on the lift and/or drop height is not required if the cask is lifted with a device designed in accordance with applicable stress limits from ANSI N14.6, and/or NUREG-0612, and has redundant drop protection features.

   b. For anchored casks, the ISFSI pad shall be designed to meet the embedment requirements of the anchorage design. A cask tip-over event for an anchored cask is not credible. The ISFSI pad shall be verified by analysis to limit cask deceleration during a design basis drop event to ≤ 45 g's at the top of the MPC fuel basket, except as provided for in this paragraph below. Analyses shall be performed using methodologies consistent with those described in the HI-STORM 100 FSAR. A restriction on the lift and/or drop height is not required to be established if the cask is lifted with a device designed in accordance with applicable stress limits from ANSI N14.6, and/or NUREG-0612, and has redundant drop protection features.

(continued)

NRC RSI 3-1:

Provide Revised Supplement Number 56, "MPC-68M Lift Lugs" in support of HI-STORM FSAR Proposed Revision 13.

Holtec Report No: HI-2012787 (Rev. 19), Table 12.0, "List of Supplements," indicates that Supplement No. 56, "MPC-68M Lift Lugs" was revised for Proposed Rev. 13 of the HI-STORM 100 FSAR. The weight of the fuel assemblies for the MPC 68-M is increased from 730 lb. to 830 lb. The demonstration of the adequacy of the capacity of the lifting lugs for this added weight has not been included in the "Structural Calculation Package for the MPC."

This information is needed to determine compliance with 10 CFR 72.236(b).

Holtec's Response to RSI 3-1:

The MPC lift lugs, shown in Holtec Licensing Drawing 3923 (Section 1.5 of HI-STORM 100 FSAR) and Holtec Fabrication Drawing 8507, are only used to lift an empty MPC without lid, i.e., in this case, the MPC-68M enclosure vessel (w/o lid) with an empty MPC-68M basket. This is consistent with the input data to Supplement No. 56, which shows the total lifted load as the sum of the enclosure vessel (w/o lid) weight (23,396 lb – 9,650 lb = 13,746 lb) plus the empty fuel basket weight (7,077 lb). Therefore, the increase in the weight of fuel assemblies from 730 lb to 830 lb does not impact the MPC-68M lift lug analysis (Supplement No. 56 of HI-2012787).

The analysis was revised to use the acceptance criteria for interfacing lift points per NUREG-0612 (1/10th of material ultimate strength) and Regulatory Guide 3.61 (1/3rd of material yield strength). This revision is in support of proposed changes in LAR 1014-11 (see pages 3-87 and 3-88, Section 3.4.3 of HI-STORM 100 FSAR, HI-2002444, Chapter 3).

In addition, changes have been made to Chapters 2 and 3 with respect to lifting criteria (consistent with changes made in pages 3-87 and 3-88, Section 3.4.3) which were mistakenly omitted in the original LAR 1014-11 submittal. These additional changes, while not directly related to RSI 3-1, are prompted by the same general update to the lifting criteria in Section 3.4.3 of the HI-STORM 100 FSAR.

NRC RSI 4-1:

Demonstrate that the proposed change #5, "To evaluate mixture of low enriched CILC fuel and normal fuel," is thermally bound by previously analyzed contents and structurers, systems, and components important to safety, and that they are within their operating temperature range.

The change did not appear to be addressed in the thermal chapter of the application.

This information is needed to determine compliance with 10 CFR 72.236(f).

Holtec Response to RSI 4-1:

As defined in FSAR Chapter 6, Section 6.III.4.4 CILC fuel is Copper Induced Locally Corroded BWR fuel. CILC fuel cladding may potentially have corrosion induced damage but does not have grossly breached spent fuel rods. Accordingly CILC fuel does not require placement in Damaged Fuel Containers (DFC) for storage. The fuel may be inter-mingled with other intact fuel assemblies for fuel storage and complies with the heat load limits specified in Chapter 2 of the HI-STORM FSAR for intact fuel. The Chapter 4 thermal analyses under normal, off-normal and accident events remains applicable to this fuel.

The above licensing basis proposed for the HI-STORM 100 application is identical to a prior request and approval of CILC fuel storage in a parallel HI-STORM FW application (Docket Number 1032, Amendment Request 1).

NRC RSI 4-2:

Demonstrate that the proposed change #8, "to increase the burnup limit to accommodate non-fuel hardware (NFH) consisting of NSA in combination with other control components," is thermally bound by previously analyzed contents and structures, systems, and components importation to safety, and that they are within their operating temperature range.

The change did not appear to be addressed in the thermal chapter of the application.

This information is needed to determine compliance with 10 CFR 72.236(f).

Holtec's Response to RSI 4-2:

The previous analysis remains thermally bounding as explained next. Increased burnup limits of non-fuel hardware does not affect permitted thermal loading in the fuel storage cells because compliance to decay heat limits must account for fuel decay heat plus any non-fuel hardware heat. This requirement is mandated in the technical specifications as quoted below:

> CoC No. 1014, Appendix B, Section 2.4.4:
> "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."

NRC RSI 4-3:

Describe in the thermal chapter of the application how the previous steady state and transient thermal analyses for normal, off-normal, and accident conditions are bounding for the inclusion of the thoria rod canister, as well as per thoria rod, as they relate to proposed changes #9, "Expand the allowable contents for the MPC-68M to include the thoria rods/canister," and #10, "Add a second permissible composition for thoria rods for all MPC-68 models." Alternatively, provide steady state and transient thermal analyses for normal, off-normal, and accident conditions that consider the cask decay heat distribution with the inclusion of the thoria rod canister.

The relatively lower decay heat of the thoria rod canister (less than or equal to 115 watts) may allow relatively higher decay heat fuel assemblies to be loaded in the cask. This could change the cask decay heat distribution and may result in higher predicted fuel and component temperatures. The application has not clearly described the thermal analysis of any potential changes in cask decay heat distribution, or how the previous analysis bounds any potential changes in cask decay heat distribution due to the relatively lower decay heat thoria rod canister. In addition, it has not been addressed if the predicted fuel or component temperatures are bounding for the decay heat per thoria rod.

This information is needed to determine compliance with 10 CFR 72.236(f).

Holtec's Response to RSI 4-3:

The application submitted does not request any change to the cask decay heat distribution for storage of the Thoria Rod Canisters in all MPC-68 models, and therefore there is no change to the bounding thermal analyses. Amendment 10 (unchanged in Amendment 11) to CoC No. 1014, Appendix B, "Approved Contents and Design Features," Table 2.4-1 provides the maximum allowable decay heat per fuel storage location for ZR-clad fuel in uniform fuel loading for each MPC model. As described in the HI-STORM 100 FSAR, with uniform fuel loading, every basket cell is assumed to be occupied with fuel producing heat at the maximum rate. This maximum decay heat applies to standard fuel assemblies as well as damaged fuel and fuel debris. Similar to damaged fuel and fuel debris, the thoria rods are stored in a canister. The maximum allowable decay heat limit for damaged fuel and fuel debris stored in an MPC-68F is 115 Watts, while the maximum allowable decay heat limit for damaged fuel and fuel debris stored in an MPC-68/68FF/68M is 393 Watts.

Conversely, the maximum decay heat for the Thoria Rod Canister is 115 Watts, as stated in CoC No. 1014, Amendment 10 (unchanged in Amendment 11), Appendix B, Table 2.1-1 (i.e., Items II.A.d and III.A.d). This limitation is significantly below the maximum allowable limit of 393 Watts for damaged fuel and fuel debris stored in an MPC-68/68M/68FF, and less than or equal to the 115 Watts maximum allowable limit for the MPC-68F. Therefore, the predicted fuel and component temperatures for a uniformly loaded MPC-68 model canister that includes the Thoria Rod Canister are bounded by the previously calculated values and no new analyses are necessary. There is no change to the previously approved decay heat distribution in any of the MPCs.

During regionalized loading, which could potentially increase the per storage location heat load limit, a greater margin to the cladding temperature limits may exist when the thoria rod canister is loaded in any MPC-68 model.

Additionally, the storage of the thoria rod canister in the MPC-68M has been approved by the staff in an exemption request (Docket No. 72-37, TAC No. L24989). The changes requested in Amendment 11 to the HI-STORM 100 are identical to that request.

NRC RSI 4-4:

Provide the following information related to proposed changes #9 and #10, respectively:

a. provide experimental data or calculations that demonstrate the best estimate hoop stress that the thoria rod fuel cladding experiences during vacuum drying is bounded by the stresses expected in UO_2 rods for the fuel that is being loaded into the MPC-68M at the maximum temperature calculated in the HI-STORM 100 FSAR; and

b. provide the same information requested in part (a.) above for the second proposed composition for thoria rods for all MPC-68 models.

The HI-STORM 100 FSAR, Table 4.III.5, "Maximum MPC-68M Temperatures Under Vacuum Drying Scenarios," shows that the maximum temperature calculated during vacuum drying is 754°F which exceeds the limit of 752°F in ISG-11, Rev. 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel." The 752°F limit was based on the stresses expected to be experienced in UO_2 based fuel. It is not clear to the NRC staff if the cladding stresses expected to be experienced in the thoria rods is bounded by that in the UO_2 rods. If the stress is greater in the thoria rods compared to the UO_2 rods, the temperature limit may be lower than 752°F. A higher short-term temperature limit may be used for low burnup thoria rod fuel if it is shown that the best estimate hoop stress that the thoria rod fuel cladding experiences is bounded by the stresses expected in UO_2 rods at the maximum temperature calculated in the HI-STORM 100 FSAR.

This information is needed to determine compliance with 10 CFR 72.236(f).

Holtec's Response to RSI 4-4:

Part a:

The DNPS Thoria Rod Canister contains 18 thoria rods which have obtained a relatively low burnup of less than 16,000 MWD/MTU. These rods were removed from two DNPS Unit 1 8x8 fuel assemblies (i.e., nine rods from each assembly). As stated in CoC No. 1014, Amendment 10, Appendix B, Table 2.1-1, "Fuel Assembly Limits" the cladding on the thoria rods is zirconium-based. The thoria rods within the Thoria Rod Canister experience similar stresses as the UO_2 rods that would be stored with the Thoria Rod Canister in a single MPC-68M. The heat load of an MPC-68M which contains the Thoria Rod Canister will be well below the design basis heat load of 36.9 kW.

Given that the expected heat load will be much lower than the design basis value, fuel and component temperatures are also expected to be well below the values specified in Table 4.III.5 of the HI-STORM 100 FSAR, Revision 12 during vacuum drying operations, and thus below the 400°C limit specified in ISG-11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," Revision 3.

Therefore, the NRC's conclusions in the May 10, 2012 SER that approved CoC No. 1014, Amendment 8 concerning hoop stress and hydride reorientation bounds the storage of the Thoria Rod Canister in an MPC-68M (emphasis added):

However, based on the staff's evaluation, it is expected that the fuel assemblies with the moderate burnup, as described in scenario (A), are not likely to have a significant amount of hydride reorientation due to limited hydride content. Furthermore, most of the low or moderate burnup fuel has hoop stresses below 90 MPa. Even if hydride reorientation occurred during storage, the network of reoriented hydrides is not expected to be extensive enough in moderate burnup fuel to cause fuel rod failures. Given the conditions of hydride reorientation and hoop stress described above and the fact that the calculated temperature is just 1°C (2°F) over the allowable limit, the staff finds the cladding temperature of 401°F in scenario A is acceptable for this application. This is based on the applicant's use of other conservative assumptions (e.g., the water in the HI-TRAC annulus is conservatively assumed to be boiling with a water temperature of 111°C (232°F) in the calculation. Additionally, the hydrostatic head of water at the annulus with the MPC bottom surface insulation causes boiling at higher than 100°C (212°F) used in the model analyses, and that the fuel rods in MPC-68M should not fail during the moisture removal operations in Scenario (A). In scenario (B), the calculated cladding temperature of 389°C (732°F) is well below the allowable limit of 400°C (752°F) as identified in ISG-11.

The staff found the evaluations of scenarios of (A) and (B) acceptable based on two conservative assumptions: (1) the water in the HI-TRAC annulus is assumed to be boiling 111°C (232°F) under the hydrostatic head of water at the annulus bottom and (2) the bottom surface of the MPC is insulated. The staff finds that the maximum cladding temperatures under scenarios (A) and (B) are in compliance with thermal limits review guidance provided in ISG-11, Rev 3.

Part b:

The thoria rod canister is approved for storage in the MPC-68, 68F and 68FF. The only change requested in this amendment to the MPC-68, 68F and 68FF models is the negligible change to the ThO_2 and UO_2 compositions, which has no impact on the hoop stress or hydride reorientation. The revised composition of the rods does not have any impact on the cladding properties. Therefore no further evaluation is required for those MPC models.

Both the storage of the thoria rods in the MPC-68M and the revised composition were both previously approved by the staff in an exemption request (Docket No. 72-37, TAC No. L24989). The changes requested in this amendment request are identical to those previously approved changes.

NRC RSI 4-5:

Provide the following information related to proposed change #4, "To provide more options on surveillance requirements and actions to be taken in the event of Overpack vents blockage when containing a loaded MPC":

a. Provide justification for the ambient temperature used in the thermal analysis or modify the thermal analysis to use an ambient temperature that is bounding for all cask locations, and considers seasonal extreme hotter temperatures that are justified for the surveillance frequency.

The application does not address an ambient temperature that is bounding for all cask locations and considers the seasonal extreme temperatures given the proposed surveillance frequency.

This information is needed to determine compliance with 10 CFR 72.236(f).

Holtec's Response to RSI 4-5:

Seasonal extreme temperatures are defined in the HI-STORM FSAR under the Off-Normal and Extreme Ambient temperature events in Table 2.2.2. These are short duration events (upto 3-days) wherein ambient temperatures excursions are postulated to reach 100°F under Off-Normal and 125°F under Extreme Ambient conditions. Due to the large thermal inertia of the HI-STORM cask these temperature variations do not affect baseline steady-state temperature profiles evaluated under the long-term 80°F Design ambient temperature. As a defense-in-depth if these temperatures are postulated to remain constant for a substantial time duration to reach steady state then to leading order HI-STORM temperature rise is reasonably estimated by the elevation in ambient temperature δ = 20°F (Off-Normal) and 45°F (Extreme Ambient)[1]. The HI-STORM temperatures under the 100% Vent Blockage computed in this manner are evaluated in Table 4-5-A. The results support the conclusion that HI-STORM component and fuel temperatures are within safety limits under seasonal extreme hotter temperature excursions.

[1] NUREG-2174 studies show that an increase in ambient temperature may result in slightly higher increase in PCT. However, in this case due to robust margins, a reasonable estimate method is acceptable.

Table 4-5-A: Maximum 100% Vent Blockage HI-STORM Temperatures Under Off-Normal and Extreme Ambient Temperatures, °F

	Normal Ambient (Note 1)	Off-Normal Ambient (Note 2)	Extreme Ambient (Note 3)	Accident Temperature Limit	Safety Criteria Compliance
Fuel Cladding	716	736	761	1058	Yes
MPC Basket	712	732	757	950	Yes
MPC Shell	482	502	527	775	Yes
Overpack Inner Shell	407	427	452	800	Yes
Overpack Concrete Section	270	290	315	350	Yes
Lid Concrete Bottom Plate	375	395	420	800	Yes
Lid Concrete Section	293	313	338	350	Yes

Note 1: Table 4.6.9 of HI-STORM 100 FSAR.
Note 2: Obtained by adding 20°F to Normal Ambient temperatures.
Note 3: Obtained by adding 45°F to Normal Ambient temperatures.

NRC RSI 4-6:

Demonstrate that operating procedures:

a. address that the revised monitoring frequency is applied only to casks with less than 19kW decay heat, and

b. address that the potential scenario where the revised monitoring frequency is applied to casks with greater than 19 kW decay heat does not cause the fuel cladding, or structures, systems, and components (SSCs) important to safety to exceed normal conditions temperature limits.

Related to proposed change #4, the application does not address that the 30-day surveillance frequency could be applied to the incorrect cask (i.e. one with a decay heat higher than 19kW). These types of errors may be challenging to detect and could result in fuel cladding temperatures exceeding the normal conditions 752°F fuel cladding temperature limit or SSCs important to safety exceeding the normal conditions temperature limits.

This information is needed to determine compliance with 10 CFR 72.236(f).

Holtec's Response to RSI 4-6:

Part a:

It should be noted that HI-STORM 100 FSAR is a high level licensing document developed to meet the requirements of 10 CFR 72. The HI-STORM 100 FSAR therefore includes high level/general operating procedures and requirements, implementation of which may vary on a site-specific basis. Therefore, the implementing procedures for cask loading, including verification and documentation of loaded assemblies and total canister heat load may vary on a site-specific basis, as will the process of ensuring that the 30-day surveillance frequency is only applied to casks with less than 19 kW decay heat.

In accordance with Section 8.1.4 of the HI-STORM 100 FSAR Revision 12, fuel assembly selection verification is performed using plant fuel records shall to ensure that only fuel assemblies that meet all CoC conditions are selected for loading into the MPC. Following loading of the pre-selected fuel assemblies into the MPC, a visual verification of the assembly identification is performed to confirm that the serial numbers match the approved fuel loading pattern. Loading operations may be aided by an underwater camera or other suitable viewing device.

The redundancy of pre and post-loading verifications of fuel assemblies, performed in accordance with Section 8.1.4 of the HI-STORM 100 FSAR, ensures that each assembly's decay heat load (and therefore the heat load of the entire canister) is known and documented prior to placement of the system on the ISFSI pad. To provide further defense-in-depth to ensure that the

30-day surveillance frequency is applied only to casks loaded with less than 19 kW of decay heat, the following note is added after Section 8.2.1 of the HI-STORM 100 FSAR Rev. 12:

"CAUTION: Some LCO requirements in the HI-STORM 100 CoC are based on individual system parameters (such as MPC total heat load). Sites should be aware of the variation in these requirements, and ensure procedures clearly identify how to implement these variations."

Part b:

As discussed in the response to "Part a", site-specific procedures are developed for loading and surveillance activities, including the requirements for surveillance frequencies. Any violation of a site-specific procedure is handled through the site's corrective action program, which includes development of any actions that need to be taken. This process is identical to the process used for the existing surveillance frequencies.

NRC RSI 4-7:

Clarify in Section 4.6.2.4 of the application that the 100% blocked vent analysis at 19kW is for normal conditions and, in addition, include the normal conditions temperature limits in Table 4.6.9 of the application. Section C.2.1 of the application should also demonstrate that the normal conditions limits should not be exceeded.

It has not been clearly stated that the 100% blocked vent analysis and associated temperature limits (e.g., 752°F for the fuel cladding and normal conditions limits for SSCs important to safety) is for normal conditions considering the length of the surveillance period.

This information is needed to determine compliance with 10 CFR 72.236(f).

Holtec's Response to RSI 4-7:

The intent of the application is to meet 100% blocked vent event to accident temperature limits. The requested 30-day surveillance period is supported by ISG 11, Rev. 3 basis for fuel cladding accident temperature limits as quoted below:

> "The basis for using 570°C is established by the creep tests conducted on irradiated Zircaloy-4 rods (Einziger, et al., 1982). The results from these experiments indicated that no cladding ruptures were observed for test times of 30 and 73 days."

Text matter in Section 4.6.2.4 is aligned with the above evaluation.

NRC RSI 4-8:

Revise Section 11.2.13.2 of the application to address that surveillance of casks is mandatory.

The application should clearly address the surveillance requirements of SSCs that are important to safety.

This information is needed to determine compliance with 10 CFR 72.236(b) and (f).

Holtec's Response to RSI 4-8:

Section 11.2.13.2 revised to accord with comment.

NRC RSI 4-9:

Provide justification and additional description for each of the following changes in the FSAR regarding how the models will be used to evaluate the event using site-specific conditions:

a. Section 4.5.2, "Time-to-Boil for a Water-Filled MPC," - "An alternate method using the FLUENT thermal model described in Section 4.5.1 can be adopted to evaluate the time for water within the MPC to boil using site-specific conditions."
b. Section 4.6.2.1(a), "HI-STORM Fire" – "An alternate method using the FLUENT thermal model described in Section 4.4 can be adopted to evaluate HI-STORM site-specific fire accident event similar to that described in Section 4.6 of HI-STORM FW FSAR."
c. Section 4.6.2.1(b), "HI-TRAC Fire," – "An alternate method using the FLUENT thermal model described in Section 4.5 can be adopted to evaluate HI-TRAC site-specific fire accident event."
d. Section 4.III.5.2, "Maximum Time Limit During Wet Transfer Operations" – "An alternate method using the FLUENT thermal model described in Section 4.III.5.1 can be adopted to evaluate the time for water within the MPC to boil using site-specific conditions."
e. Section 4.III.6.2(a)(i), "HI-STORM Fire" – "An alternate method using the FLUENT thermal model described in Section 4.III.4 can be adopted to evaluate HI-STORM site-specific fire accident event similar to that described in Section 4.6 of HI-STORM FW FSAR."
f. Section 4.III.6.2(a)(ii), "HI-TRAC Fire," – "An alternate method using the FLUENT thermal model described in Section 4.III.5 can be adopted to evaluate HI-TRAC site-specific fire accident event."

It is not clear how the thermal models will be used to evaluate the fire or time-to-boil using site-specific conditions. In addition, these changes were not included in the summary of proposed changes.

This information is needed to determine compliance with 10 CFR 72.236(f).

Holtec's Response to RSI 4-9:

Suitable methodologies addressing site specific evaluation of time-to-boil and HI-TRAC/HI-STORM fires is incorporated in FSAR Chapter 4, Sections 4.5.2, 4.6.2, 4.III.5.2 and 4.III.6.2.

welds to the MPC lid and shell, as discussed in Section 2.2.4. In addition, the threaded holes in the MPC lid are designed in accordance with the requirements of ~~ANSI N14.6~~NUREG-0612 and Regulatory Guide 3.61 for critical lifts to facilitate vertical MPC transfer.

Helium leakage testing of the MPC base metals (shell, baseplate, and MPC lid) and MPC shell to baseplate and shell to shell welds is performed on the unloaded MPC.

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

The structural analysis of the MPC, in conjunction with the redundant closures and nondestructive examination, pressure testing, and helium leak testing, (performed on the vent and drain port cover plates), provides assurance of canister closure integrity in lieu of the specific weld joint requirements of Section III, Subsection NB.

Compliance with the ASME Code as it is applied to the design and fabrication of the MPC and the associated justification are discussed in Section 2.2.4. The MPC is designed for all design basis normal, off-normal, and postulated accident conditions, as defined in Section 2.2. These design loadings include postulated drop accidents while in the cavity of the HI-STORM overpack or the HI-TRAC transfer cask. The load combinations for which the MPC is designed are defined in Section 2.2.7. The maximum allowable weight and dimensions of a fuel assembly to be stored in the MPC are limited in accordance with Section 2.1.5.

Thermal

The design and operation of the HI-STORM 100 System meets the intent of the review guidance contained in ISG-11, Revision 3 [2.0.8]. Specifically, the ISG-11 provisions that are explicitly invoked and satisfied are:

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

ii. The maximum value of the *calculated* temperature for all CSF (including ZR and stainless steel fuel cladding materials) under long-term normal conditions of storage must remain below 400°C (752°F). For short-term operations, including canister drying, helium backfill, and on-site cask transport operations, the fuel cladding temperature must not exceed 400°C (752°F) for high burnup fuel and 570°C (1058°F) for moderate burnup fuel.

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

overpack. Effluents generated during MPC loading and closure operations are handled by the plant's radwaste system and procedures under the licensee's 10CFR50 license.

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

Acceptance Tests and Maintenance

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

Decommissioning

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

2.0.3 HI-TRAC Transfer Cask Design Criteria

General

The HI-TRAC transfer cask is designed for 40 years of service, while satisfying the requirements of 10CFR72. The adequacy of the HI-TRAC design for the design life is discussed in Section 3.4.11.

Structural

The HI-TRAC transfer cask includes both structural and non-structural biological shielding components that are classified as important to safety. The structural steel components of the HI-TRAC, with the exception of the lifting trunnions, are designed and fabricated in accordance with the applicable requirements of Section III, Subsection NF, of the ASME Code with certain NRC-approved alternatives, as discussed in Section 2.2.4. The lifting trunnions and associated attachments are designed in accordance with the requirements of NUREG-0612 and ~~ANSI N14.6~~Regulatory Guide 3.61 for non-redundant lifting devices.

The HI-TRAC transfer cask is designed for all normal, off-normal, and design basis accident condition loadings, as defined in Section 2.2. At a minimum, the HI-TRAC transfer cask must protect the MPC from deformation, provide continued adequate performance, and allow the retrieval of the MPC under all conditions. These design loadings include a side drop from the maximum allowable handling height, consistent with the technical specifications. The load combinations for which the HI-TRAC is designed are defined in Section 2.2.7. The physical characteristics of each

Table 2.0.1
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Life:			
Design	40 yrs.	-	Table 1.2.2
License	20 yrs.	10CFR72.42(a) and 10CFR72.236(g)	-
Structural:			
Design Codes:			
Enclosure Vessel	ASME Code, Section III, Subsection NB	10CFR72.24(c)(4)	Section 2.0.1
Fuel Basket	ASME Code, Section III, Subsection NG for core supports (NG-1121)	10CFR72.24(c)(4)	Section 2.0.1
MPC Fuel Basket Supports (Angled Plates)	ASME Code, Section III, Subsection NG for internal structures (NG-1122)	10CFR72.24(c)(4)	Section 2.0.1
MPC Lifting Points	~~ANSI N14.6/~~NUREG-0612 & Regulatory Guide 3.61	10CFR72.24(c)(4)	Section 1.2.1.4
Dead Weights†:			
Max. Loaded Canister (dry)	90,000 lb.	R.G. 3.61	Table 3.2.1
Empty Canister (dry)	42,000 lb. (MPC-24) 45,000 lb. (MPC-24E/EF) 39,000 lb. (MPC-68/68F/68FF) 36,000 lb. (MPC-32)	R.G. 3.61	Table 3.2.1
Design Cavity Pressures:			
Normal:	100 psig	ANSI/ANS 57.9	Section 2.2.1.3
Off-Normal:	110 psig	ANSI/ANS 57.9	Section 2.2.2.1
Accident (Internal)	200 psig	ANSI/ANS 57.9	Section 2.2.3.8

† Weights listed in this table are bounding weights. Actual weights will be less, and will vary based on as-built dimensions of the components, fuel type, and the presence of fuel spacers and non-fuel hardware.

Table 2.0.1 (continued)
MPC DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Handling Loads	115% of Dead Weight	CMAA #70	Section 2.2.1.2
Lifting Attachment Acceptance Criteria	1/10 Ultimate 1/~~6~~ 3 Yield	NUREG-0612 Regulatory Guide 3.61~~ANSI N14.6~~	Section 3.4.3
Attachment/Component Interface Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Section 3.4.3
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3
Wet/Dry Loading	Wet or Dry	-	Section 1.2.2.2
Transfer Orientation	Vertical	-	Section 1.2.2.2
Storage Orientation	Vertical	-	Section 1.2.2.2
Fuel Rod Rupture Releases:			
Source Term Release Fraction	1%	NUREG-1536	Sections 2.2.1.3
Fill Gases	100%	NUREG-1536	Sections 2.2.1.3
Fission Gases	30%	NUREG-1536	Sections 2.2.1.3
Snow and Ice	Protected by Overpack	ASCE 7-88	Section 2.2.1.6
Off-Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Temperature	See Tables 2.0.2 and 2.0.3	ANSI/ANS 57.9	Section 2.2.2.2
Leakage of One Seal	N/A	ISG-18	Sections 2.2.2.4 and 7.1
Partial Blockage of Overpack Air Inlets	50% of Air Inlets Blocked	-	Section 2.2.2.5
Source Term Release Fraction:			
Fuel Rod Failures	10%	NUREG-1536	Sections 2.2.2.1
Fill Gases	100%	NUREG-1536	Sections 2.2.2.1
Fission Gases	30%	NUREG-1536	Sections 2.2.2.1
Design-Basis (Postulated) Accident Design Events and Conditions:		10CFR72.24(d)(2) & 10CFR72.94	
Tip Over	See Table 2.0.2	-	Section 2.2.3.2
End Drop	See Table 2.0.2	-	Section 2.2.3.1

Table 2.0.2 (continued)
HI-STORM 100 OVERPACK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Beyond Controlled Area During Normal Operation and Anticipated Occurrences	25 mrem/yr. to whole body 75 mrem/yr. to thyroid 25 mrem/yr. to any critical organ	10CFR72.104	Sections 5.1.1, 7.2, and 10.1
At Controlled Area Boundary from Design Basis Accident	5 rem TEDE or sum of DDE and CDE to any individual organ or tissue (other than lens of eye) ≤ 50 rem. 15 rem lens dose. 50 rem shallow dose to skin or extremity.	10CFR72.106	Sections 5.1.2, 7.3, and 10.1
Design Bases:			
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1.9
Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Outside Temperatures:			
Max. Yearly Average	80° F	ANSI/ANS 57.9	Section 2.2.1.4
Live Load†:		ANSI/ANS 57.9	-
Loaded Transfer Cask (max.)	250,000 lb. (HI-TRAC 125 w/transfer lid)	R.G. 3.61	Table 3.2.4 Section 2.2.1.2
Dry Loaded MPC (max.)	90,000 lb.	R.G. 3.61	Table 3.2.1 and Section 2.2.1.2
Handling:			Section 2.2.1.2
Handling Loads	115% of Dead Weight	CMAA #70	Section 2.2.1.2
Lifting ~~Attachment~~ Acceptance Criteria	~~1/10 Ultimate~~ 1/~~6~~ 3 Yield	~~NUREG-0612~~ Regulatory Guide 3.61~~ANSI N14.6~~	Section 3.4.3.5
~~Attachment/Component Interface Acceptance Criteria~~	~~1/3 Yield~~	~~Regulatory Guide 3.61~~	~~Section 3.4.3~~
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3

† Weights listed in this table are bounding weights. Actual weights will be less, and will vary based on as-built dimensions of the components, fuel type, and the presence of fuel spacers and non-fuel hardware, as applicable.

TABLE 2.0.3
HI-TRAC TRANSFER CASK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Life:			
Design	40 yrs.	-	Section 2.0.3
License	20 yrs.	10CFR72.42(a) & 10CFR72.236(g)	
Structural:			
Design Codes:			
Structural Steel	ASME Code, Section III, Subsection NF	10CFR72.24(c)(4)	Section 2.0.3
Lifting Trunnions	NUREG-0612 & ~~ANSI N14.6~~Regulatory Guide 3.61	10CFR72.24(c)(4)	Section 1.2.1.4
Dead Weights[†]:			
Max. Empty Cask:			
w/top lid and pool lid installed and water jacket filled	143,500 lb. (HI-TRAC 125) 102,000 lb. (HI-TRAC 100) 102,000 lb. (HI-TRAC 100D) 146,000 lb. (HI-TRAC 125D)	R.G. 3.61	Table 3.2.2
w/top lid and transfer lid installed and water jacket filled (N/A for HI-TRAC 100D and 125D)	155,000 lb. (HI-TRAC 125) 111,000 lb. (HI-TRAC 100)	R.G. 3.61	Table 3.2.2
Max. MPC/HI-TRAC with Yoke (in-pool lift):	250,000 lb. (HI-TRAC 125 and 125D) 200,000 lb. (HI-TRAC 100 and 100D)	R.G. 3.61	Table 3.2.4
Design Cavity Pressures:			
HI-TRAC Cavity	Hydrostatic	ANSI/ANS 57.9	Section 2.2.1.3

[†] Weights listed in this table are bounding weights. Actual weights will be less, and will vary based on as-built dimensions of the components, fuel type, and the presence of fuel spacers and non-fuel hardware, as applicable.

TABLE 2.0.3 (continued)
HI-TRAC TRANSFER CASK DESIGN CRITERIA SUMMARY

Type	Criteria	Basis	FSAR Reference
Design Bases:			
Spent Fuel Specification	See Table 2.0.1	10CFR72.236(a)	Section 2.1
Normal Design Event Conditions:		10CFR72.122(b)(1)	
Ambient Temperature:	80 ° F	ANSI/ANS 57.9	Section 2.2.1.4
Live Load†			
Max. Loaded Canister			
Dry	90,000 lb.	R.G. 3.61	Table 3.2.1
Wet (including water in HI-TRAC annulus)	106,570 lb.	R.G. 3.61	Table 3.2.4
Handling:			Section 2.2.1.2
Handling Loads	115% of Dead Weight	CMAA #70	Section 2.2.1.2
Lifting Attachment Acceptance Criteria	1/10 Ultimate 1/~~6~~ 3 Yield	NUREG-0612 Regulatory Guide 3.61~~ANSI N14.6~~	Section 3.4.3
Attachment/Component Interface Acceptance Criteria	1/3 Yield	Regulatory Guide 3.61	Section 3.4.3
Away from Attachment Acceptance Criteria	ASME Code Level A	ASME Code	Section 3.4.3
Minimum Temperature for Handling Operations	0° F	ANSI/ANS 57.9	Section 2.2.1.2
Wet/Dry Loading	Wet or Dry	-	Section 1.2.2.2
Transfer Orientation	Vertical	-	Section 1.2.2.2
Test Loads:			
Trunnions	300% of vertical design load	NUREG-0612 & ~~ANSI N14.6~~Regulatory Guide 3.61	Section 9.1.2.1
Design-Basis (Postulated) Accident Design Events and Conditions:		10CFR72.24(d)(2) & 10CFR72.94	
Side Drop	42 in.	-	Section 2.2.3.1

† Weights listed in this table are bounding weights. Actual weights will be less, and will vary based on as-built dimensions of the components, fuel type, and the presence of fuel spacers and non-fuel hardware, as applicable.

determined to be acceptable for storage in the HI-STORM 100 System. Section 2.1.9 summarizes the authorized contents for the HI-STORM 100 System. Any fuel assembly that has fuel characteristics within the range of Tables 2.1.3 and 2.1.4 and meets the other limits specified in Section 2.1.9 is acceptable for storage in the HI-STORM 100 System. Tables 2.1.3 and 2.1.4 present the groups of fuel assembly types defined as "array/classes" as described in further detail in Chapter 6. Table 2.1.5 lists the BWR and PWR fuel assembly designs which are found to govern for three qualification criteria, namely reactivity, shielding, and thermal. Additional information on the design basis fuel definition is presented in the following subsections.

2.1.2 Intact SNF Specifications

Intact fuel assemblies are defined as fuel assemblies without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. The design payload for the HI-STORM 100 System is intact ZR or stainless steel (SS) clad fuel assemblies with the characteristics listed in Tables 2.1.17 through 2.1.24.

Intact fuel assemblies without fuel rods in fuel rod locations cannot be loaded into the HI-STORM 100 unless dummy fuel rods, which occupy a volume greater than or equal to the original fuel rods, replace the missing rods prior to loading. Any intact fuel assembly that falls within the geometric, thermal, and nuclear limits established for the design basis intact fuel assembly, as defined in Section 2.1.9 can be safely stored in the HI-STORM 100 System. If irradiated dummy stainless steel rods are present in the fuel assembly, the dummy/replacement rods will be considered in the site specific dose calculations.

The range of fuel characteristics specified in Tables 2.1.3 and 2.1.4 have been evaluated in this FSAR and are acceptable for storage in the HI-STORM 100 System within the decay heat, burnup, and cooling time limits specified in Section 2.1.9 for intact fuel assemblies.

2.1.3 Damaged SNF and Fuel Debris Specifications

Damaged fuel and fuel debris are defined in Table 1.0.1.

Damaged fuel assemblies and fuel debris will be loaded into stainless steel damaged fuel containers (DFCs) provided with mesh screens having between 40x40 and 250x250 openings per inch, for storage in the HI-STORM 100 System (see Figures 2.1.1 and 2.1.2B, C, and D). The MPC-24, MPC-24EF, MPC-32 and MPC-32F are designed to accommodate PWR damaged fuel and fuel debris. The MPC-68, MPC-68F and MPC-68FF are designed to accommodate BWR damaged fuel and fuel debris. The appropriate structural, thermal, shielding, criticality, and confinement analyses 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 Section 2.1.9. Dresden Unit 1 fuel assemblies contained in Transnuclear-designed damaged fuel canisters and one Dresden Unit 1 thoria rod canister have been approved for storage directly in the HI-STORM 100 System without re-packaging (see Figures 2.1.2 and 2.1.2A).

and Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source have been qualified for storage. Up to one Thoria Rod Canister is authorized for storage in combination with other intact and damaged fuel, and fuel debris as specified in Section 2.1.9.

Non-fuel hardware, as defined in Table 1.0.1, has been evaluated and is authorized for storage in the PWR MPCs as specified in Section 2.1.9.

2.1.8 Criticality Parameters for Design Basis SNF

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

The minimum ^{10}B areal density in the neutron absorber panels for each MPC model is shown in Table 2.1.15.

For all MPCs, the ^{10}B areal density used for the criticality analysis is conservatively established below the minimum values shown in Table 2.1.15. For Boral, the value used in the analysis is 75% of the minimum value, while for METAMIC, it is 90% of the minimum value. This is consistent with NUREG-1536 [2.1.5] which suggests a 25% reduction in ^{10}B areal density credit when subject to standard acceptance tests, and which allows a smaller reduction when more comprehensive tests of the areal density are performed.

The criticality analyses for the MPC-24, MPC-24E and MPC-24EF (all with higher enriched fuel) and for the MPC-32 and MPC-32F were performed with credit taken for soluble boron in the MPC water during wet loading and unloading operations. Table 2.1.14 and 2.1.16 provide the required soluble boron concentrations for these MPCs.

2.1.9 Summary of Authorized Contents

Tables 2.1.3, 2.1.4, 2.1.12, and 2.1.17 through 2.1.2~~9~~7 together specify the limits for spent fuel and non-fuel hardware authorized for storage in the HI-STORM 100 System. The limits in these tables are derived from the safety analyses described in the following chapters of this FSAR. Fuel classified as damaged fuel assemblies or fuel debris must be stored in damaged fuel containers for storage in the HI-STORM 100 System.

Tables 2.1.17 through 2.1.24 are the baseline tables that specify the fuel assembly limits for each of the MPC models, with appropriate references to the other tables in this section for certain other limits. Tables 2.1.17 through 2.1.24 refer to Section 2.1.9.1 for ZR-clad fuel limits on minimum cooling time, maximum decay heat, and maximum burnup for uniform and regionalized fuel loading.

2.1.9.1 Decay Heat, Burnup, and Cooling Time Limits for ZR-Clad Fuel

Each ZR-clad fuel assembly and any PWR integral non-fuel hardware (NFH) to be stored in the HI-STORM 100 System must meet the following limits, in addition to meeting the physical limits specified elsewhere in this section, to be authorized for storage in the HI-STORM 100 System. The contents of each fuel storage location (fuel assembly and NFH) to be stored must be verified to have, as applicable:

- A decay heat less than or equal to the maximum allowable value.
- ~~An assembly average enrichment greater than or equal to the minimum value used in determining the maximum allowable burnup.~~
- A burnup less than or equal to the maximum allowable value.
- A cooling time greater than or equal to the minimum allowable value.

The maximum allowable ZR-clad fuel storage location decay heat values are determined using the methodology described in Section 2.1.9.1.1 or 2.1.9.1.2 depending on whether uniform fuel loading or regionalized fuel loading is being implemented†. The total permissible MPC heat load, for both uniform and regionalized loading, is determined in the following two subsections 2.1.9.1.1 and 2.1.9.1.2. The decay heat limits are independent of burnup, cooling time, or enrichment and are based strictly on the thermal analysis described in Chapter 4. Decay heat limits must be met for all contents in a fuel storage location (i.e., fuel and PWR non-fuel hardware, as applicable).

~~The maximum allowable average burnup per fuel storage location is determined by calculation as a function of minimum enrichment, maximum allowable decay heat, and minimum cooling time from 3 to 20 years, as described in Section 2.1.9.1.3.~~

~~Section 12.2 describes how compliance with these limits may be verified, including practical examples.~~

2.1.9.1.1 <u>Uniform Fuel Loading Decay Heat Limits for ZR-Clad Fuel</u>

Table 2.1.26 provides the maximum allowable decay heat per fuel storage location for ZR-clad fuel in uniform fuel loading for each MPC model in aboveground storage*. Even if the limits in Table 2.1.26 are met, the user must follow the instructions in the next section to calculate Q_{CoC} to determine if certain operational steps are required per the CoC. If the user needs to load fuel assemblies with a decay heat higher than the limits in Table 2.1.26, a regionalized loading pattern discussed in the next section may be considered.

† Note that the stainless steel-clad fuel decay heat limits apply to all fuel in the MPC, if a mixture of stainless steel and ZR-clad fuel is stored in the same MPC. The stainless steel-clad fuel assembly decay heat limits may be found in Table 2.1.17 through 2.1.24

* Maximum allowable heat loads in 100U underground storage are defined in Supplement 2.I; however the discussion in Section 2.1.9.1 also applies to the 100U.

$$q_1 = X \times q_2 \qquad \text{Equation i}$$

Using the steps provided above we find for X=2 that q_1 = 1.43 kW and q_2 = 0.715 kW for MPC-32. The user can follow Table 2.1.30 for discrete values of X to determine q_1 and q_2 or calculate q_1 and q_2 for a specific value of X using the steps above. It should be noted that equation e is used to determine Q_{CoC} when following the heat load limits for regionalized loading.

It should be emphasized that the variable two-region scheme of storage does not introduce any new complication in the dry storage implementation. As compared to uniform loading in MPC-32, where q = 1.0625 kW for all cells, the regionalized loading gives the user the flexibility to load the MPC with more varying heat loads. It is noted that for X < 1 Q_{CoC} is greater than Q_d, for X = 1 Q_{CoC} equals Q_d, and for X >1 Q_{CoC} is less than Q_d. For ALARA and regardless of which loading pattern is used, a plant should always seek to preferentially locate the fuel with the higher heat loads toward the center of the MPC. If the need arises to place younger fuel into dry storage a regionalized pattern with X <1 may be more appropriate.

2.1.9.1.3 DeletedBurnup Limits as a Function of Cooling Time for ZR-Clad Fuel

The maximum allowable ZR-clad fuel assembly average burnup varies with the following parameters, based on the shielding analysis in Chapter 5:

- Minimum required fuel assembly cooling time
- Maximum allowable fuel assembly decay heat
- Minimum fuel assembly average enrichment

The calculation described in this section is used to determine the maximum allowable fuel assembly burnup for minimum cooling times between 3 and 20 years, using maximum decay heat and minimum enrichment as input values. This calculation may be used to create multiple burnup versus cooling time tables for a particular fuel assembly array/class and different minimum enrichments. The allowable maximum burnup for a specific fuel assembly may be calculated based on the assembly's particular enrichment and cooling time.

(i) Choose a fuel assembly minimum enrichment, E_{235}.

(ii) Calculate the maximum allowable fuel assembly average burnup for a minimum cooling time between 3 and 20 years using the equation below:

$$Bu = (A \times q) + (B \times q^2) + (C \times q^3) + [D \times (E_{235})^2] + (E \times q \times E_{235}) + (F \times q^2 \times E_{235}) + G$$

Equation j

Where:

Bu = Maximum allowable assembly average burnup (MWD/MTU)

q = Maximum allowable decay heat per fuel storage location determined in Section 2.1.9.1.1 or 2.1.9.1.2 (kW)

E_{235} = Minimum fuel assembly average enrichment (wt. % ^{235}U) (e.g., for 4.05 wt. %, use 4.05)

A through G = Coefficients from Tables 2.1.28 or 2.1.29 for the applicable fuel assembly array/class and minimum cooling time.

2.1.9.1.4 DeletedOther Considerations

In computing the allowable maximum fuel storage location decay heats and fuel assembly average burnups, the following requirements apply:

- Calculated burnup limits shall be rounded down to the nearest integer
- Calculated burnup limits greater than 68,200 MWD/MTU for PWR fuel and 65,000 MWD/MTU for BWR fuel must be reduced to be equal to these values.
- Linear interpolation of calculated burnups between cooling times for a given fuel assembly maximum decay heat and minimum enrichment is permitted. For example, the allowable burnup for a minimum cooling time of 4.5 years may be interpolated between those burnups calculated for 4 and 5 years.
- ZR-clad fuel assemblies must have a minimum enrichment, as defined in Table 1.0.1, greater than or equal to the value used in determining the maximum allowable burnup per Section 2.1.9.1.3 to be authorized for storage in the MPC.
- 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 PWR 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.

Section 12.2.10 provides a practical example of determining fuel storage location decay heat, burnup, and cooling time limits and verifying compliance for a set of example fuel assemblies.

2.1.9.1.5 Supplemental Cooling Threshold Heat Loads

Fuel loading operations involving the handling of High Burnup Fuel (HBF) in a dewatered MPC emplaced in a HI-TRAC transfer cask require additional cooling under certain thermal loads to address reduced heat dissipation relative to the normal storage condition. To address this requirement

Table 2.1.5

DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION

Criterion	**BWR Fuel**	**PWR Fuel**
Reactivity (Criticality)	GE12/14 10x10 with Partial Length Rods (Array/Class 10x10A)	B&W 15x15 (Array/Class 15x15F)
Shielding	GE 7x7	B&W 15x15
Thermal-Hydraulic	GE-12/14 10x10	W 17x17 OFA
Structural	730 Lb for in-tact fuel and 830 Lb for canisterized fuel (in-tact and canisterized fuel include channels)	1680 Lb including any control components

Table 2.1.12

DESIGN CHARACTERISTICS FOR THORIA RODS IN D-1 THORIA ROD CANISTERS

PARAMETER	MPC-68 or MPC-68F
Cladding Type	Zircaloy
Composition	98.2 wt.% ThO_2, 1.8 wt.% UO_2 with an enrichment of 93.5 wt. % ^{235}U or 98.5 wt.% ThO_2, 1.5 wt.% UO_2 with an enrichment of 93.5 wt. % ^{235}U
Number of Rods Per Thoria Canister	$\leq$ 18
Decay Heat Per Thoria Canister	$\leq$ 115 watts
Post-Irradiation Fuel Cooling Time and Average Burnup Per Thoria Canister	Cooling time $\geq$ 18 years and average burnup $\leq$ 16,000 MWD/MTIHM
Initial Heavy Metal Weight	$\leq$ 27 kg/canister
Fuel Cladding O.D.	$\geq$ 0.412 inches
Fuel Cladding I.D.	$\leq$ 0.362 inches
Fuel Pellet O.D.	$\leq$ 0.358 inches
Active Fuel Length	$\leq$ 111 inches
Canister Weight	$\leq$ 550 lbs., including Thoria Rods
Canister Material	Type 304 SS

Table 2.1.16

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

Fuel Assembly Array/Class	All Intact Fuel Assemblies		One or More Damaged Fuel Assemblies or Fuel Debris	
	Max. Initial Enrichment ≤ 4.1 wt.% ^{235}U (ppmb)	**Max. Initial Enrichment 5.0 wt.% ^{235}U (ppmb)**	**Max. Initial Enrichment ≤ 4.1 wt.% ^{235}U (ppmb)**	**Max. Initial Enrichment 5.0 wt.% ^{235}U (ppmb)**
14x14A/B/C/D/E	1,300	1,900	1,500	2,300
15x15A/B/C/G/I	1,800	2,500	1,900	2,700
15x15D/E/F/H	1,900	2,600	2,100	2,900
16x16A	1,400	2,000	1,500	2,300
17x17A	1,600	2,200	1,800	2,600
17x17~~A/~~B/C	1,900	2,600	2,100	2,900

Note:

1. For maximum initial enrichments between 4.1 wt% and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be determined by linear interpolation between the minimum soluble boron concentrations at 4.1 wt% and 5.~~1~~0 wt% ^{235}U.

Table 2.1.17

LIMITS FOR MATERIAL TO BE STORED IN MPC-24

PARAMETER	VALUE
Fuel Type	Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class
Cladding Type	ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3 for the applicable array/class
Post-irradiation Cooling Time and Average Burnup per Assembly	ZR clad: ≥ 3 years and ≤ 68,200 MWD/MTU~~As specified in Section 2.1.9.1~~ SS clad: ≥ 8 years and ≤ 40,000 MWD/MTU
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1 SS clad: ≤ 710 Watts
Non-Fuel Hardware Burnup and Cooling Time	As specified in Table 2.1.25
Fuel Assembly Length	≤ 176.8 in. (nominal design)
Fuel Assembly Width	≤ 8.54 in. (nominal design)
Fuel Assembly Weight	≤ 1,720 lbs (including non-fuel hardware) for array/classes that do not require fuel spacers, otherwise ≤ 1,680 lbs (including non-fuel hardware)
Other Limitations	▪ Quantity is limited to up to 24 PWR intact fuel assemblies. ▪ Damaged fuel assemblies and fuel debris are not permitted for loading in MPC-24. ▪ One NSA is authorized to be loaded with a fuel assembly in fuel storage location 9, 10, 15, or 16. ▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts, with or without ITTRs, may be stored with fuel assemblies in any fuel cell location. ▪ APSRs may be loaded with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16 ▪ CRAs, RCCAs and/or CEAs may be stored with fuel assemblies in fuel cell locations 4, 5, 8 through 11, 14 through 17, 20, and/or 21. ▪ Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.

Table 2.1.20

LIMITS FOR MATERIAL TO BE STORED IN MPC-24E AND MPC-24EF

PARAMETER	VALUE (Note 1)	
Fuel Type	Uranium oxide PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class	Uranium oxide PWR damaged fuel assemblies and/or fuel debris meeting the limits in Table 2.1.3 for the applicable array/class, placed in a Damaged Fuel Container (DFC)
Cladding Type	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3 for the applicable array/class	As specified in Table 2.1.3 for the applicable array/class
Post-irradiation Cooling Time, and Average Burnup per Assembly	ZR clad: ≥ 3 yrs and ≤ 68,200 MWD/MTU~~As specified in Section 2.1.9.1~~ SS clad: ≥ 8 yrs and ≤ 40,000 MWD/MTU	ZR clad: ≥ 3 yrs and ≤ 68,200 MWD/MTU~~As specified in Section 2.1.9.1~~ SS clad: ≥ 8 yrs and ≤ 40,000 MWD/MTU
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1 SS clad: ≤ 710 Watts	ZR clad: As specified in Section 2.1.9.1 SS clad: ≤ 710 Watts
Non-fuel hardware post-irradiation Cooling Time and Burnup	As specified in Table 2.1.25	As specified in Table 2.1.25
Fuel Assembly Length	≤ 176.8 in. (nominal design)	≤ 176.8 in. (nominal design)
Fuel Assembly Width	≤ 8.54 in. (nominal design)	≤ 8.54 in. (nominal design)
Fuel Assembly Weight	≤ 1,720 lbs (including non-fuel hardware) for array/classes that do not require fuel spacers, otherwise ≤ 1680 lbs (including non-fuel hardware)	≤ 1,720 lbs (including DFC and non-fuel hardware) for array/classes that do not require fuel spacers, otherwise ≤ 1680 lbs (including DFC and non-fuel hardware)

Table 2.1.22

LIMITS FOR MATERIAL TO BE STORED IN MPC-68 AND MPC-68FF

PARAMETER	VALUE (Note 1)	
Fuel Type	Uranium oxide or MOX BWR intact fuel assemblies meeting the limits in Table 2.1.4 for the applicable array/class, with or without channels.	Uranium oxide or MOX BWR damaged fuel assemblies or fuel debris meeting the limits in Table 2.1.4 for the applicable array/class, with or without channels, in DFCs.
Cladding Type	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.4 for the applicable array/class	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.4 for the applicable array/class
Maximum Initial Planar Average Enrichment per Assembly and Rod Enrichment	As specified in Table 2.1.4 for the applicable fuel assembly array/class	Planar Average: $\leq$ 2.7 wt% ^{235}U for array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A; $\leq$ 4.0 wt% ^{235}U for all other array/classes Rod: As specified in Table 2.1.4
Post-irradiation cooling time and average burnup per Assembly	ZR clad: $\geq$ 3 yrs and $\leq$ 65,000 MWD/MTU~~As specified in Section 2.1.9.1~~; except as provided in Notes 2 and 3. SS clad: Note 4	ZR clad: $\geq$ 3 yrs and $\leq$ 65,000 MWD/MTU~~As specified in Section 2.1.9.1~~; except as provided in Notes 2 and 3. SS clad: Note 4.
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: $\leq$ 95 Watts	ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: $\leq$ 95 Watts
Fuel Assembly Length	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: $\leq$ 135.0 in. (nominal design) All Other array/classes: $\leq$ 176.5 in. (nominal design)	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: $\leq$ 135.0 in. (nominal design) All Other array/classes: $\leq$ 176.5 in. (nominal design)

Table 2.1.22 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68 AND MPC-68FF

PARAMETER	VALUE (Note 1)	
Fuel Assembly Width	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 4.7 in. (nominal design) All Other array/classes: ≤ 5.85 in. (nominal design)	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 4.7 in. (nominal design) All Other array/classes: ≤ 5.85 in. (nominal design)
Fuel Assembly Weight	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 400 lbs. (including channels) All Other array/classes: ≤ 730 lbs. (including channels)	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 550 lbs. (including channels and DFC) All Other array/classes: ≤ ~~7~~830 lbs. (including channels and DFC)
Other Limitations	▪ For assembly/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A, up to 68 intact fuel assemblies or damaged fuel assemblies in DFCs may be stored. Fuel debris in DFCs may be stored in up to 8 locations. A Dresden Unit 1 Thoria Rod Container may be stored in one location. ▪ For all other array/classes, up to 16 DFCs containing damaged fuel assemblies and/or up to eight (8) DFCs containing fuel assemblies classified as fuel debris may be stored. DFCs shall be located only in fuel cell locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68, with the balance comprised of intact fuel assemblies meeting the above specifications, up to a total of 68. ▪ SS-clad fuel assemblies with stainless steel channels must be stored in fuel cell locations 19 through 22, 28 through 31, 38 through 41, and/or 47 through 50. ▪ Dresden Unit 1 fuel assemblies with one antimony-beryllium neutron source are permitted. The antimony-beryllium neutron source material shall be in a water rod location.	

NOTES:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.
2. Array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies shall have a cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU or MWD/MTIHM, and a decay heat ≤ 115 Watts.
3. Array/class 8x8F fuel assemblies shall have a cooling time ≥ 10 years, an average burnup ≤ 27,500 MWD/MTU, and a decay heat ≤ 183.5 Watts.
4. SS-clad fuel assemblies shall have a cooling time ≥ 10 years, and an average burnup ≤ 22,500 MWD/MTU.

Table 2.1.24

LIMITS FOR MATERIAL TO BE STORED IN MPC-32 AND MPC-32F

PARAMETER	**VALUE (Note 1)**	
Fuel Type	Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class	Uranium oxide, PWR damaged fuel assemblies and fuel debris in DFCs meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class
Cladding Type	ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable fuel assembly array/class	ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable fuel assembly array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3	As specified in Table 2.1.3
Post-irradiation Cooling Time and Average Burnup per Assembly	ZR clad: ≥ 3 yrs and ≤ 68,200 MWD/MTU~~As specified in Section 2.1.9.1~~ SS clad: ≥ 9 years and ≤ 30,000 MWD/MTU or ≥ 20 years and ≤ 40,000MWD/MTU	ZR clad: ≥ 3 yrs and ≤ 68,200 MWD/MTU~~As specified in Section 2.1.9.1~~ SS clad: ≥ 9 years and ≤ 30,000 MWD/MTU or ≥ 20 years and ≤ 40,000MWD/MTU
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1 SS clad: ≤ 500 Watts	ZR clad: As specified in Section 2.1.9.1 SS clad: ≤ 500 Watts
Non-fuel hardware post-irradiation Cooling Time and Burnup	As specified in Table 2.1.25	As specified in Table 2.1.25
Fuel Assembly Length	≤ 176.8 in. (nominal design)	≤ 176.8 in. (nominal design)
Fuel Assembly Width	≤ 8.54 in. (nominal design)	≤ 8.54 in. (nominal design)
Fuel Assembly Weight	≤ 1,720 lbs (including non-fuel hardware) for array/classes that do not require fuel spacers, otherwise ≤ 1,680 lbs (including non-fuel hardware)	≤ 1,720 lbs (including DFC and non-fuel hardware) for array/classes that do not require fuel spacers, otherwise ≤ 1,680 lbs (including DFC and non-fuel hardware)

Table 2.1.25

NON-FUEL HARDWARE BURNUP AND COOLING TIME LIMITS (Notes 1, 2, 3, and 87)

Post-irradiation Cooling Time (yrs)	NSA with NFH, Inserts (Note 4) Maximum Burnup (MWD/MTU)	NSA or Guide Tube Hardware (Note 5) Maximum Burnup (MWD/MTU)	NSA without NFH, Guide Tube Hardware, or Control Component (Note 65) Maximum Burnup (MWD/MTU)	APSR Maximum Burnup (MWD/MTU)
≥ 3	≤ 24,635	N/A (Note 7)	N/A (Note 6)	N/A
≥ 4	≤ 30,000	≤ 20,000	N/A	N/A
≥ 5	≤ 36,748	≤ 25,000	≤ 630,000	≤ 45,000
≥ 6	≤ 44,102	≤ 30,000	-	≤ 54,500
≥ 7	≤ 52,900	≤ 40,000	-	≤ 68,000
≥ 8	≤ 60,000	≤ 45,000	-	≤ 83,000
≥ 9	≤ 78,784-	≤ 50,000	-	≤ 111,000
≥ 10	≤ 101,826-	≤ 60,000	-	≤ 180,000
≥ 11	≤ 141,982-	≤ 75,000	-	≤ 630,000
≥ 12	≤ 360,000-	≤ 90,000	-	-
≥ 13	-	≤ 180,000	-	-
≥ 14	-	≤ 630,000	-	-

NOTES:

1. 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.
2. Linear interpolation between points is permitted, except that NSA or Guide Tube Hardware and APSR burnups > 180,000 MWD/MTU and ≤ 630,000 MWD/MTU must be cooled ≥ 14 years and ≥ 11 years, respectively.
3. Applicable to uniform loading and regionalized loading.
4. Includes Burnable Poison Rod Assemblies (BPRAs), Wet Annular Burnable Absorbers (WABAs), and vibration suppressor inserts.and Neutron Source Assemblies (NSAs) in combination with other control components (i.e. BPRAs, TPDs, and/or RCCAs).
5. Includes Thimble Plug Devices (TPDs), water displacement guide tube plugs, and orifice rod assemblies, Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), Rod Cluster Control Assemblies (RCCAs) and NSAs without other forms of control

components.

6. Includes Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), and Rod Cluster Control Assemblies (RCCAs).

76. N/A means not authorized for loading at this cooling time.

87. Non-fuel hardware burnup and cooling time limits are not applicable to Instrument Tube Tie Rods (ITTRs), since they are installed post-irradiation.

Table 2.1.28

[INTENTIONALLY DELETED]

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 14x14A						
	A	B	C	D	E	F	G
≥3	19311.5	275.367	-59.0252	-139.41	2851.12	-451.845	-615.413
≥4	33865.9	-5473.03	851.121	-132.739	3408.58	-656.479	-609.523
≥5	46686.2	-13226.9	2588.39	-150.149	3871.87	-806.533	-90.2065
≥6	56328.9	-20443.2	4547.38	-176.815	4299.19	-927.358	603.192
≥7	64136	-27137.5	6628.18	-200.933	4669.22	-1018.94	797.162
≥8	71744.1	-34290.3	9036.9	-214.249	4886.95	-1037.59	508.703
≥9	77262	-39724.2	11061	-228.2	5141.35	-1102.05	338.294
≥10	82939.8	-45575.6	13320.2	-233.691	5266.25	-1095.94	-73.3159
≥11	86541	-49289.6	14921.7	-242.092	5444.54	-1141.6	-83.0603
≥12	91383	-54456.7	17107	-242.881	5528.7	-1149.2	-547.579
≥13	95877.6	-59404.7	19268	-240.36	5524.35	-1094.72	-933.64
≥14	97648.3	-61091.6	20261.7	-244.234	5654.56	-1151.47	-749.836
≥15	102533	-66651.5	22799.7	-240.858	5647.05	-1120.32	-1293.34
≥16	106216	-70753.8	24830.1	-237.04	5647.63	-1099.12	-1583.89
≥17	109863	-75005	27038	-234.299	5652.45	-1080.98	-1862.07
≥18	111460	-76482.3	28076.5	-234.426	5703.52	-1104.39	-1695.77
≥19	114916	-80339.6	30126.5	-229.73	5663.21	-1065.48	-1941.83
≥20	119592	-86161.5	33258.2	-227.256	5700.49	-1100.21	-2474.01

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 14x14B						
	A	B	C	D	E	F	G
≥3	18036.1	63.7639	-24.7251	-130.732	2449.87	-347.748	-858.192
≥4	30303.4	-4304.2	598.79	-118.757	2853.18	-486.453	-459.902
≥5	40779.6	-9922.93	1722.83	-138.174	3255.69	-608.267	245.251
≥6	48806.7	-15248.9	3021.47	-158.69	3570.24	-689.876	833.917
≥7	55070.5	-19934.6	4325.62	-179.964	3870.33	-765.849	1203.89
≥8	60619.6	-24346	5649.29	-189.701	4042.23	-795.324	1158.12
≥9	64605.7	-27677.1	6778.12	-205.459	4292.35	-877.966	1169.88
≥10	69083.8	-31509.4	8072.42	-206.157	4358.01	-875.041	856.449
≥11	72663.2	-34663.9	9228.96	-209.199	4442.68	-889.512	671.567
≥12	74808.9	-36367	9948.88	-214.344	4571.29	-942.418	765.261
≥13	78340.3	-39541.1	11173.8	-212.8	4615.06	-957.833	410.807
≥14	81274.8	-42172.3	12259.9	-209.758	4626.13	-958.016	190.59
≥15	83961.4	-44624.5	13329.1	-207.697	4632.16	-952.876	20.8575
≥16	84968.5	-44982.1	13615.8	-207.171	4683.41	-992.162	247.54
≥17	87721.6	-47543.1	14781.4	-203.373	4674.3	-988.577	37.9689
≥18	90562.9	-50100.4	15940.4	-198.649	4651.64	-982.459	-247.421
≥19	93011.6	-52316.6	17049.9	-194.964	4644.76	-994.63	-413.021
≥20	95567.8	-54566.6	18124	-190.22	4593.92	-963.412	-551.983

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 14x14C						
	A	B	C	D	E	F	G
≥3	18263.7	174.161	-57.6694	-138.112	2539.74	-369.764	-1372.33
≥4	30514.5	-4291.52	562.37	-124.944	2869.17	-481.139	-889.883
≥5	41338	-10325.7	1752.96	-141.247	3146.48	-535.709	-248.078
≥6	48969.7	-15421.3	2966.33	-163.574	3429.74	-587.225	429.331
≥7	55384.6	-20228.9	4261.47	-180.846	3654.55	-617.255	599.251
≥8	60240.2	-24093.2	5418.86	-199.974	3893.72	-663.995	693.934
≥9	64729	-27745.7	6545.45	-205.385	3986.06	-650.124	512.528
≥10	68413.7	-30942.2	7651.29	-216.408	4174.71	-702.931	380.431
≥11	71870.6	-33906.7	8692.81	-218.813	4248.28	-704.458	160.645
≥12	74918.4	-36522	9660.01	-218.248	4283.68	-696.498	-29.0682
≥13	77348.3	-38613.7	10501.8	-220.644	4348.23	-702.266	-118.646
≥14	79817.1	-40661.8	11331.2	-218.711	4382.32	-710.578	-236.123
≥15	82354.2	-42858.3	12257.3	-215.835	4405.89	-718.805	-431.051
≥16	84787.2	-44994.5	13185.9	-213.386	4410.99	-711.437	-572.104
≥17	87084.6	-46866.1	14004.8	-206.788	4360.3	-679.542	-724.721
≥18	88083.1	-47387.1	14393.4	-208.681	4420.85	-709.311	-534.454
≥19	90783.6	-49760.6	15462.7	-203.649	4403.3	-705.741	-773.066
≥20	93212	-51753.3	16401.5	-197.232	4361.65	-692.925	-964.628

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 15x15A/B/C						
	A	B	C	D	E	F	G
≥3	15037.3	108.689	-18.8378	-127.422	2050.02	-242.828	-580.66
≥4	25506.6	-2994.03	356.834	-116.45	2430.25	-350.901	-356.378
≥5	34788.8	-7173.07	1065.9	-124.785	2712.23	-424.681	267.705
≥6	41948.6	-11225.3	1912.12	-145.727	3003.29	-489.538	852.112
≥7	47524.9	-14770.9	2755.16	-165.889	3253.9	-542.7	1146.96
≥8	52596.9	-18348.8	3699.72	-177.17	3415.69	-567.012	1021.41
≥9	56055.4	-20837.1	4430.93	-192.168	3625.93	-623.325	1058.61
≥10	59611.3	-23402.1	5179.52	-195.105	3699.18	-626.448	868.517
≥11	62765.3	-25766.5	5924.71	-195.57	3749.91	-627.139	667.124
≥12	65664.4	-28004.8	6670.75	-195.08	3788.33	-628.904	410.783
≥13	67281.7	-29116.7	7120.59	-202.817	3929.38	-688.738	492.309
≥14	69961.4	-31158.6	7834.02	-197.988	3917.29	-677.565	266.561
≥15	72146	-32795.7	8453.67	-195.083	3931.47	-681.037	99.0606
≥16	74142.6	-34244.8	9023.57	-190.645	3905.54	-663.682	10.8885
≥17	76411.4	-36026.3	9729.98	-188.874	3911.21	-663.449	-151.805
≥18	77091	-36088	9884.09	-188.554	3965.08	-708.55	59.3839
≥19	79194.5	-37566.4	10477.5	-181.656	3906.93	-682.4	-117.952
≥20	81600.4	-39464.5	11281.9	-175.182	3869.49	-677.179	-367.705

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 15x15D/E/F/H/I						
	A	B	C	D	E	F	G
≥3	14376.7	102.205	-20.6279	-126.017	1903.36	-210.883	-493.065
≥4	24351.4	-2686.57	297.975	-110.819	2233.78	-301.615	-152.713
≥5	33518.4	-6711.35	958.544	-122.85	2522.7	-371.286	392.608
≥6	40377	-10472.4	1718.53	-144.535	2793.29	-426.436	951.528
≥7	46105.8	-13996.2	2515.32	-157.827	2962.46	-445.314	1100.56
≥8	50219.7	-16677.7	3198.3	-175.057	3176.74	-492.727	1223.62
≥9	54281.2	-19555.6	3983.47	-181.703	3279.03	-499.997	1034.55
≥10	56761.6	-21287.3	4525.98	-195.045	3470.41	-559.074	1103.3
≥11	59820	-23445.2	5165.43	-194.997	3518.23	-561.422	862.68
≥12	62287.2	-25164.6	5709.9	-194.771	3552.69	-561.466	680.488
≥13	64799	-27023.7	6335.16	-192.121	3570.41	-561.326	469.583
≥14	66938.7	-28593.1	6892.63	-194.226	3632.92	-583.997	319.867
≥15	68116.5	-29148.6	7140.09	-192.545	3670.39	-607.278	395.344
≥16	70154.9	-30570.1	7662.91	-187.366	3649.14	-597.205	232.318
≥17	72042.5	-31867.6	8169.01	-183.453	3646.92	-603.907	96.0388
≥18	73719.8	-32926.1	8596.12	-177.896	3614.57	-592.868	46.6774
≥19	75183.1	-33727.4	8949.64	-172.386	3581.13	-586.347	3.57256
≥20	77306.1	-35449	9690.02	-173.784	3636.87	-626.321	-205.513

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 16x16A						
	A	B	C	D	E	F	G
≥3	16226.8	143.714	-32.4809	-136.707	2255.33	-291.683	-699.947
≥4	27844.2	-3590.69	444.838	-124.301	2644.09	-411.598	-381.106
≥5	38191.5	-8678.48	1361.58	-132.855	2910.45	-473.183	224.473
≥6	46382.2	-13819.6	2511.32	-158.262	3216.92	-532.337	706.656
≥7	52692.3	-18289	3657.18	-179.765	3488.3	-583.133	908.839
≥8	57758.7	-22133.7	4736.88	-199.014	3717.42	-618.83	944.903
≥9	62363.3	-25798.7	5841.18	-207.025	3844.38	-625.741	734.928
≥10	66659.1	-29416.3	6993.31	-216.458	3981.97	-642.641	389.366
≥11	69262.7	-31452.7	7724.66	-220.836	4107.55	-681.043	407.121
≥12	72631.5	-34291.9	8704.8	-219.929	4131.5	-662.513	100.093
≥13	75375.3	-36589.3	9555.88	-217.994	4143.15	-644.014	-62.3294
≥14	78178.7	-39097.1	10532	-221.923	4226.28	-667.012	-317.743
≥15	79706.3	-40104	10993.3	-218.751	4242.12	-670.665	-205.579
≥16	82392.6	-42418.9	11940.7	-216.278	4274.09	-689.236	-479.752
≥17	84521.8	-44150.5	12683.3	-212.056	4245.99	-665.418	-558.901
≥18	86777.1	-45984.8	13479	-204.867	4180.8	-621.805	-716.366
≥19	89179.7	-48109.8	14434.5	-206.484	4230.03	-648.557	-902.1
≥20	90141.7	-48401.4	14702.6	-203.284	4245.54	-670.655	-734.604

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 17x17A/16x16B/C						
	A	B	C	D	E	F	G
≥3	15985.1	3.53963	-9.04955	-128.835	2149.5	-260.415	-262.997
≥4	27532.9	-3494.41	428.199	-119.504	2603.01	-390.91	-140.319
≥5	38481.2	-8870.98	1411.03	-139.279	3008.46	-492.881	388.377
≥6	47410.9	-14479.6	2679.08	-162.13	3335.48	-557.777	702.164
≥7	54596.8	-19703.2	4043.46	-181.339	3586.06	-587.634	804.05
≥8	60146.1	-24003.4	5271.54	-201.262	3830.32	-621.706	848.454
≥9	65006.3	-27951	6479.04	-210.753	3977.69	-627.805	615.84
≥10	69216	-31614.7	7712.58	-222.423	4173.4	-672.33	387.879
≥11	73001.3	-34871.1	8824.44	-225.128	4238.28	-657.259	101.654
≥12	76326.1	-37795.9	9887.35	-226.731	4298.11	-647.55	-122.236
≥13	78859.9	-40058.9	10797.1	-231.798	4402.14	-669.982	-203.383
≥14	82201.3	-43032.5	11934.1	-228.162	4417.99	-661.61	-561.969
≥15	84950	-45544.6	12972.4	-225.369	4417.84	-637.422	-771.254
≥16	87511.8	-47720	13857.7	-219.255	4365.24	-585.655	-907.775
≥17	90496.4	-50728.9	15186	-223.019	4446.51	-613.378	-1200.94
≥18	91392.5	-51002.4	15461.4	-220.272	4475.28	-636.398	-1003.81
≥19	94343.9	-53670.8	16631.6	-214.045	4441.31	-616.201	-1310.01
≥20	96562.9	-55591.2	17553.4	-209.917	4397.67	-573.199	-1380.64

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 17x17B/C						
	A	B	C	D	E	F	G
≥3	14738	47.5402	-13.8187	-127.895	1946.58	-219.289	-389.029
≥4	25285.2	-3011.92	350.116	-115.75	2316.89	-319.23	-220.413
≥5	34589.6	-7130.34	1037.26	-128.673	2627.27	-394.58	459.642
≥6	42056.2	-11353.7	1908.68	-150.234	2897.38	-444.316	923.971
≥7	47977.6	-15204.8	2827.4	-173.349	3178.25	-504.16	1138.82
≥8	52924	-18547.6	3671.08	-183.025	3298.64	-501.278	1064.68
≥9	56465.5	-21139.4	4435.67	-200.386	3538	-569.712	1078.78
≥10	60190.9	-23872.7	5224.31	-203.233	3602.88	-562.312	805.336
≥11	63482.1	-26431.1	6035.79	-205.096	3668.84	-566.889	536.011
≥12	66095	-28311.8	6637.72	-204.367	3692.68	-555.305	372.223
≥13	67757.4	-29474.4	7094.08	-211.649	3826.42	-606.886	437.412
≥14	70403.7	-31517.4	7807.15	-207.668	3828.69	-601.081	183.09
≥15	72506.5	-33036.1	8372.59	-203.428	3823.38	-594.995	47.5175
≥16	74625.2	-34620.5	8974.32	-199.003	3798.57	-573.098	-95.0221
≥17	76549	-35952.6	9498.14	-193.459	3766.52	-556.928	-190.662
≥18	77871.9	-36785.5	9916.91	-195.592	3837.65	-599.45	-152.261
≥19	79834.8	-38191.6	10501.9	-190.83	3812.46	-589.635	-286.847
≥20	81975.5	-39777.2	11174.5	-185.767	3795.78	-595.664	-475.978

Table 2.1.29

[INTENTIONALLY DELETED]

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 7x7B & 10x10F‡						
	A	B	C	D	E	F	G
≥3	26409.1	28347.5	-16858	-147.076	5636.32	-1606.75	1177.88
≥4	61967.8	-6618.31	-4131.96	-113.949	6122.77	-2042.85	-96.7439
≥5	91601.1	-49298.3	17826.5	-132.045	6823.14	-2418.49	-185.189
≥6	111369	-80890.1	35713.8	-150.262	7288.51	-2471.1	86.6363
≥7	126904	-108669	53338.1	-167.764	7650.57	-2340.78	150.403
≥8	139181	-132294	69852.5	-187.317	8098.66	-2336.13	97.5285
≥9	150334	-154490	86148.1	-193.899	8232.84	-2040.37	-123.029
≥10	159897	-173614	100819	-194.156	8254.99	-1708.32	-373.605
≥11	166931	-186860	111502	-193.776	8251.55	-1393.91	-543.677
≥12	173691	-201687	125166	-202.578	8626.84	-1642.3	-650.814
≥13	180312	-215406	137518	-201.041	8642.19	-1469.45	-810.024
≥14	185927	-227005	148721	-197.938	8607.6	-1225.95	-892.876
≥15	191151	-236120	156781	-191.625	8451.86	-846.27	-1019.4
≥16	195761	-244598	165372	-187.043	8359.19	-572.561	-1068.19
≥17	200791	-256573	179816	-197.26	8914.28	-1393.37	-1218.63
≥18	206068	-266136	188841	-187.191	8569.56	-730.898	-1363.79
≥19	210187	-273609	197794	-182.151	8488.23	-584.727	-1335.59
≥20	213731	-278120	203074	-175.864	8395.63	-457.304	-1364.38

‡Array/Class 10x10F for MPC-68M only.

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 8x8B						
	A	B	C	D	E	F	G
≥3	28219.6	28963.7	-17616.2	-147.68	5887.41	-1730.96	1048.21
≥4	66061.8	-10742.4	-1961.82	-123.066	6565.54	-2356.05	-298.005
≥5	95790.7	-53401.7	19836.7	-134.584	7145.41	-2637.09	-298.858
≥6	117477	-90055.9	41383.9	-154.758	7613.43	-2612.69	-64.9921
≥7	134090	-120643	60983	-168.675	7809	-2183.3	-40.8885
≥8	148186	-149181	81418.7	-185.726	8190.07	-2040.31	-260.773
≥9	159082	-172081	99175.2	-197.185	8450.86	-1792.04	-381.705
≥10	168816	-191389	113810	-195.613	8359.87	-1244.22	-613.594
≥11	177221	-210599	131099	-208.3	8810	-1466.49	-819.773
≥12	183929	-224384	143405	-207.497	8841.33	-1227.71	-929.708
≥13	191093	-240384	158327	-204.95	8760.17	-811.708	-1154.76
≥14	196787	-252211	169664	-204.574	8810.95	-610.928	-1208.97
≥15	203345	-267656	186057	-208.962	9078.41	-828.954	-1383.76
≥16	207973	-276838	196071	-204.592	9024.17	-640.808	-1436.43
≥17	213891	-290411	211145	-202.169	9024.19	-482.1	-1595.28
≥18	217483	-294066	214600	-194.243	8859.35	-244.684	-1529.61
≥19	220504	-297897	219704	-190.161	8794.97	-10.9863	-1433.86
≥20	227821	-318395	245322	-194.682	9060.96	-350.308	-1741.16

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 8x8C/D/E						
	A	B	C	D	E	F	G
≥3	28592.7	28691.5	-17773.6	-149.418	5969.45	-1746.07	1063.62
≥4	66720.8	-12115.7	-1154	-128.444	6787.16	-2529.99	-302.155
≥5	96929.1	-55827.5	21140.3	-136.228	7259.19	-2685.06	-334.328
≥6	118190	-92000.2	42602.5	-162.204	7907.46	-2853.42	-47.5465
≥7	135120	-123437	62827.1	-172.397	8059.72	-2385.81	-75.0053
≥8	149162	-152986	84543.1	-195.458	8559.11	-2306.54	-183.595
≥9	161041	-177511	103020	-200.087	8632.84	-1864.4	-433.081
≥10	171754	-201468	122929	-209.799	8952.06	-1802.86	-755.742
≥11	179364	-217723	137000	-215.803	9142.37	-1664.82	-847.268
≥12	186090	-232150	150255	-216.033	9218.36	-1441.92	-975.817
≥13	193571	-249160	165997	-213.204	9146.99	-1011.13	-1119.47
≥14	200034	-263671	180359	-210.559	9107.54	-694.626	-1312.55
≥15	205581	-275904	193585	-216.242	9446.57	-1040.65	-1428.13
≥16	212015	-290101	207594	-210.036	9212.93	-428.321	-1590.7
≥17	216775	-299399	218278	-204.611	9187.86	-398.353	-1657.6
≥18	220653	-306719	227133	-202.498	9186.34	-181.672	-1611.86
≥19	224859	-314004	235956	-193.902	8990.14	145.151	-1604.71
≥20	228541	-320787	245449	-200.727	9310.87	-230.252	-1570.18

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9A						
	A	B	C	D	E	F	G
≥3	30538.7	28463.2	-18105.5	-150.039	6226.92	-1876.69	1034.06
≥4	71040.1	-16692.2	1164.15	-128.241	7105.27	-2728.58	-414.09
≥5	100888	-60277.7	24150.1	-142.541	7896.11	-3272.86	-232.197
≥6	124846	-102954	50350.8	-161.849	8350.16	-3163.44	-91.1396
≥7	143516	-140615	76456.5	-185.538	8833.04	-2949.38	-104.802
≥8	158218	-171718	99788.2	-196.315	9048.88	-2529.26	-259.929
≥9	172226	-204312	126620	-214.214	9511.56	-2459.19	-624.954
≥10	182700	-227938	146736	-215.793	9555.41	-1959.92	-830.943
≥11	190734	-246174	163557	-218.071	9649.43	-1647.5	-935.021
≥12	199997	-269577	186406	-223.975	9884.92	-1534.34	-1235.27
≥13	207414	-287446	204723	-228.808	10131.7	-1614.49	-1358.61
≥14	215263	-306131	223440	-220.919	9928.27	-988.276	-1638.05
≥15	221920	-321612	239503	-217.949	9839.02	-554.709	-1784.04
≥16	226532	-331778	252234	-216.189	9893.43	-442.149	-1754.72
≥17	232959	-348593	272609	-219.907	10126.3	-663.84	-1915.3
≥18	240810	-369085	296809	-219.729	10294.6	-859.302	-2218.87
≥19	244637	-375057	304456	-210.997	10077.8	-425.446	-2127.83
≥20	248112	-379262	309391	-204.191	9863.67	100.27	-2059.39

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9B						
	A	B	C	D	E	F	G
≥3	30613.2	28985.3	-18371	-151.117	6321.55	-1881.28	988.92
≥4	71346.6	-15922.9	631.132	-128.876	7232.47	-2810.64	-471.737
≥5	102131	-60654.1	23762.7	-140.748	7881.6	-3156.38	-417.979
≥6	127187	-105842	51525.2	-162.228	8307.4	-2913.08	-342.13
≥7	146853	-145834	79146.5	-185.192	8718.74	-2529.57	-484.885
≥8	162013	-178244	103205	-197.825	8896.39	-1921.58	-584.013
≥9	176764	-212856	131577	-215.41	9328.18	-1737.12	-1041.11
≥10	186900	-235819	151238	-218.98	9388.08	-1179.87	-1202.83
≥11	196178	-257688	171031	-220.323	9408.47	-638.53	-1385.16
≥12	205366	-280266	192775	-223.715	9592.12	-472.261	-1661.6
≥13	215012	-306103	218866	-231.821	9853.37	-361.449	-1985.56
≥14	222368	-324558	238655	-228.062	9834.57	3.47358	-2178.84
≥15	226705	-332738	247316	-224.659	9696.59	632.172	-2090.75
≥16	233846	-349835	265676	-221.533	9649.93	913.747	-2243.34
≥17	243979	-379622	300077	-222.351	9792.17	1011.04	-2753.36
≥18	247774	-386203	308873	-220.306	9791.37	1164.58	-2612.25
≥19	254041	-401906	327901	-213.96	9645.47	1664.94	-2786.2
≥20	256003	-402034	330566	-215.242	9850.42	1359.46	-2550.06

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9C/D						
	A	B	C	D	E	F	G
≥3	30051.6	29548.7	-18614.2	-148.276	6148.44	-1810.34	1006
≥4	70472.7	-14696.6	-233.567	-127.728	7008.69	-2634.22	-444.373
≥5	101298	-59638.9	23065.2	-138.523	7627.57	-2958.03	-377.965
≥6	125546	-102740	49217.4	-160.811	8096.34	-2798.88	-259.767
≥7	143887	-139261	74100.4	-184.302	8550.86	-2517.19	-275.151
≥8	159633	-172741	98641.4	-194.351	8636.89	-1838.81	-486.731
≥9	173517	-204709	124803	-212.604	9151.98	-1853.27	-887.137
≥10	182895	-225481	142362	-218.251	9262.59	-1408.25	-978.356
≥11	192530	-247839	162173	-217.381	9213.58	-818.676	-1222.12
≥12	201127	-268201	181030	-215.552	9147.44	-232.221	-1481.55
≥13	209538	-289761	203291	-225.092	9588.12	-574.227	-1749.35
≥14	216798	-306958	220468	-222.578	9518.22	-69.9307	-1919.71
≥15	223515	-323254	237933	-217.398	9366.52	475.506	-2012.93
≥16	228796	-334529	250541	-215.004	9369.33	662.325	-2122.75
≥17	237256	-356311	273419	-206.483	9029.55	1551.3	-2367.96
≥18	242778	-369493	290354	-215.557	9600.71	659.297	-2589.32
≥19	246704	-377971	302630	-210.768	9509.41	1025.34	-2476.06
≥20	249944	-382059	308281	-205.495	9362.63	1389.71	-2350.49

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9E/F						
	A	B	C	D	E	F	G
≥3	30284.3	26949.5	-16926.4	-147.914	6017.02	-1854.81	1026.15
≥4	69727.4	-17117.2	1982.33	-127.983	6874.68	-2673.01	-359.962
≥5	98438.9	-58492	23382.2	-138.712	7513.55	-3038.23	-112.641
≥6	119765	-95024.1	45261	-159.669	8074.25	-3129.49	221.182
≥7	136740	-128219	67940.1	-182.439	8595.68	-3098.17	315.544
≥8	150745	-156607	88691.5	-193.941	8908.73	-2947.64	142.072
≥9	162915	-182667	109134	-198.37	8999.11	-2531	-93.4908
≥10	174000	-208668	131543	-210.777	9365.52	-2511.74	-445.876
≥11	181524	-224252	145280	-212.407	9489.67	-2387.49	-544.123
≥12	188946	-240952	160787	-210.65	9478.1	-2029.94	-652.339
≥13	193762	-250900	171363	-215.798	9742.31	-2179.24	-608.636
≥14	203288	-275191	196115	-218.113	9992.5	-2437.71	-1065.92
≥15	208108	-284395	205221	-213.956	9857.25	-1970.65	-1082.94
≥16	215093	-301828	224757	-209.736	9789.58	-1718.37	-1303.35
≥17	220056	-310906	234180	-201.494	9541.73	-1230.42	-1284.15
≥18	224545	-320969	247724	-206.807	9892.97	-1790.61	-1381.9
≥19	226901	-322168	250395	-204.073	9902.14	-1748.78	-1253.22
≥20	235561	-345414	276856	-198.306	9720.78	-1284.14	-1569.18

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 9x9G						
	A	B	C	D	E	F	G
≥3	35158.5	26918.5	-17976.7	-149.915	6787.19	-2154.29	836.894
≥4	77137.2	-19760.1	2371.28	-130.934	8015.43	-3512.38	-455.424
≥5	113405	-77931.2	35511.2	-150.637	8932.55	-4099.48	-629.806
≥6	139938	-128700	68698.3	-173.799	9451.22	-3847.83	-455.905
≥7	164267	-183309	109526	-193.952	9737.91	-3046.84	-737.992
≥8	182646	-227630	146275	-210.936	10092.3	-2489.3	-1066.96
≥9	199309	-270496	184230	-218.617	10124.3	-1453.81	-1381.41
≥10	213186	-308612	221699	-235.828	10703.2	-1483.31	-1821.73
≥11	225587	-342892	256242	-236.112	10658.5	-612.076	-2134.65
≥12	235725	-370471	285195	-234.378	10604.9	118.591	-2417.89
≥13	247043	-404028	323049	-245.79	11158.2	-281.813	-2869.82
≥14	253649	-421134	342682	-243.142	11082.3	400.019	-2903.88
≥15	262750	-448593	376340	-245.435	11241.2	581.355	-3125.07
≥16	270816	-470846	402249	-236.294	10845.4	1791.46	-3293.07
≥17	279840	-500272	441964	-241.324	11222.6	1455.84	-3528.25
≥18	284533	-511287	458538	-240.905	11367.2	1459.68	-3520.94
≥19	295787	-545885	501824	-235.685	11188.2	2082.21	-3954.2
≥20	300209	-556936	519174	-229.539	10956	2942.09	-3872.87

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time (years)	Array/Class 10x10A/B/G‡						
	A	B	C	D	E	F	G
≥3	29285.4	27562.2	-16985	-148.415	5960.56	-1810.79	1001.45
≥4	67844.9	-14383	395.619	-127.723	6754.56	-2547.96	-369.267
≥5	96660.5	-55383.8	21180.4	-137.17	7296.6	-2793.58	-192.85
≥6	118098	-91995	42958	-162.985	7931.44	-2940.84	60.9197
≥7	135115	-123721	63588.9	-171.747	8060.23	-2485.59	73.6219
≥8	148721	-151690	84143.9	-190.26	8515.81	-2444.25	-63.4649
≥9	160770	-177397	104069	-197.534	8673.6	-2101.25	-331.046
≥10	170331	-198419	121817	-213.692	9178.33	-2351.54	-472.844
≥11	179130	-217799	138652	-209.75	9095.43	-1842.88	-705.254
≥12	186070	-232389	151792	-208.946	9104.52	-1565.11	-822.73
≥13	192407	-246005	164928	-209.696	9234.7	-1541.54	-979.245
≥14	200493	-265596	183851	-207.639	9159.83	-1095.72	-1240.61
≥15	205594	-276161	195760	-213.491	9564.23	-1672.22	-1333.64
≥16	209386	-282942	204110	-209.322	9515.83	-1506.86	-1286.82
≥17	214972	-295149	217095	-202.445	9292.34	-893.6	-1364.97
≥18	219312	-302748	225826	-198.667	9272.27	-878.536	-1379.58
≥19	223481	-310663	235908	-194.825	9252.9	-785.066	-1379.62
≥20	227628	-319115	247597	-199.194	9509.02	-1135.23	-1386.19

‡Array/Class 10x10G for MPC-68M only.

~~Table 2.1.29 (cont'd)~~

~~BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS (ZR-CLAD FUEL)~~

~~Cooling Time (years)~~	~~Array/Class 10x10C~~						
	~~A~~	~~B~~	~~C~~	~~D~~	~~E~~	~~F~~	~~G~~
~~≥3~~	~~31425.3~~	~~27358.9~~	~~-17413.3~~	~~-152.096~~	~~6367.53~~	~~-1967.91~~	~~925.763~~
~~≥4~~	~~71804~~	~~-16964.1~~	~~1000.4~~	~~-129.299~~	~~7227.18~~	~~-2806.44~~	~~-416.92~~
~~≥5~~	~~102685~~	~~-62383.3~~	~~24971.2~~	~~-142.316~~	~~7961~~	~~-3290.98~~	~~-354.784~~
~~≥6~~	~~126962~~	~~-105802~~	~~51444.6~~	~~-164.283~~	~~8421.44~~	~~-3104.21~~	~~-186.615~~
~~≥7~~	~~146284~~	~~-145608~~	~~79275.5~~	~~-188.967~~	~~8927.23~~	~~-2859.08~~	~~-251.163~~
~~≥8~~	~~162748~~	~~-181259~~	~~105859~~	~~-199.122~~	~~9052.91~~	~~-2206.31~~	~~-554.124~~
~~≥9~~	~~176612~~	~~-214183~~	~~133261~~	~~-217.56~~	~~9492.17~~	~~-1999.28~~	~~-860.669~~
~~≥10~~	~~187756~~	~~-239944~~	~~155315~~	~~-219.56~~	~~9532.45~~	~~-1470.9~~	~~-1113.42~~
~~≥11~~	~~196580~~	~~-260941~~	~~174536~~	~~-222.457~~	~~9591.64~~	~~-944.473~~	~~-1225.79~~
~~≥12~~	~~208017~~	~~-291492~~	~~204805~~	~~-233.488~~	~~10058.3~~	~~-1217.01~~	~~-1749.84~~
~~≥13~~	~~214920~~	~~-307772~~	~~221158~~	~~-234.747~~	~~10137.1~~	~~-897.23~~	~~-1868.04~~
~~≥14~~	~~222562~~	~~-326471~~	~~240234~~	~~-228.569~~	~~9929.34~~	~~-183.47~~	~~-2016.12~~
~~≥15~~	~~228844~~	~~-342382~~	~~258347~~	~~-226.944~~	~~9936.76~~	~~117.061~~	~~-2106.05~~
~~≥16~~	~~233907~~	~~-353008~~	~~270390~~	~~-223.179~~	~~9910.72~~	~~360.39~~	~~-2105.23~~
~~≥17~~	~~244153~~	~~-383017~~	~~304819~~	~~-227.266~~	~~10103.2~~	~~380.393~~	~~-2633.23~~
~~≥18~~	~~249240~~	~~-395456~~	~~321452~~	~~-226.989~~	~~10284.1~~	~~169.947~~	~~-2623.67~~
~~≥19~~	~~254343~~	~~-406555~~	~~335240~~	~~-220.569~~	~~10070.5~~	~~764.689~~	~~-2640.2~~
~~≥20~~	~~260202~~	~~-421069~~	~~354249~~	~~-216.255~~	~~10069.9~~	~~854.497~~	~~-2732.77~~

2.2.1 Normal Condition Design Criteria

2.2.1.1 Dead Weight

The HI-STORM 100 System must withstand the static loads due to the weights of each of its components, including the weight of the HI-TRAC with the loaded MPC atop the storage overpack.

2.2.1.2 Handling

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

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

The loads shall be increased by 15% to include any dynamic effects from the lifting operations as directed by CMAA #70 [2.2.16].

Handling operations of the loaded HI-TRAC transfer cask or HI-STORM overpack are limited to working area ambient temperatures greater than or equal to 0°F. This limitation is specified to ensure that a sufficient safety margin exists before brittle fracture might occur during handling operations. Subsection 3.1.2.3 provides the demonstration of the adequacy of the HI-TRAC transfer cask and the HI-STORM overpack for use during handling operations at a minimum service temperature of 0° F.

~~Lifting attachments and s~~Special lifting devices shall meet the requirements of ANSI N14.6† [2.2.3].

2.2.1.3 Pressure

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

The normal condition MPC internal design pressure bounds the cumulative effects of the maximum fill gas volume, normal environmental ambient temperatures, the maximum MPC heat load, and an assumed 1% of the fuel rods ruptured with 100% of the fill gas and 30% of the significant

† Yield and ultimate strength values used in the stress compliance demonstration per ANSI N14.6 shall utilize confirmed material test data through either independent coupon testing or material suppliers= CMTR or COC, as appropriate. To ensure consistency between the design and fabrication of a lifting component, compliance with ANSI N14.6 in this FSAR implies that the guidelines of ASME Section III, Subsection NF for Class 3 structures are followed for material procurement and testing, fabrication, and for NDE during manufacturing.

The HI-STORM 100 System must withstand loads due to a handling accident. Even though the loaded HI-STORM 100 System will be lifted in accordance with approved, written procedures and may use special lifting devices~~-~~ which comply~~ies~~ with ANSI N14.6-1993 [2.2.3], certain drop events are considered herein to demonstrate the defense-in-depth features of the design.

The loaded HI-STORM overpack will be lifted so that the bottom of the cask is at a height less than the vertical lift limit (see Table 2.2.8) above the ground. For conservatism, the postulated drop event assumes that the loaded HI-STORM 100 overpack falls freely from the vertical lift limit height before impacting a thick reinforced concrete pad. The deceleration of the cask must be maintained below 45 g's. Additionally, the overpack must continue to suitably shield the radiation emitted from the loaded MPC. The use of special lifting devices designed in accordance with ANSI N14.6 having redundant drop protection features to lift the loaded overpack will eliminate the lift height limit. The lift height limit is dependent on the characteristics of the impacting surface, which are specified in Table 2.2.9, and the number of fuel assemblies being stored (full or partially loaded MPC). For site-specific conditions including impact surfaces not encompassed by Table 2.2.9 or handling a partially loaded MPC, the licensee shall evaluate the site-specific conditions to ensure that the drop accident loads do not exceed 45 g's. The methodology used in this alternative analysis shall be commensurate with the analyses in Appendix 3.A and shall be reviewed by the Certificate Holder.

The loaded HI-TRAC will be lifted so that the lowest point on the transfer cask (i.e., the bottom edge of the cask/lid assemblage) is at a height less than the calculated horizontal lift height limit (see Table 2.2.8) above the ground, when lifted horizontally outside of the reactor facility. For conservatism, the postulated drop event assumes that the loaded HI-TRAC falls freely from the horizontal lift height limit before impact.

Analysis is provided that demonstrates that the HI-TRAC continues to suitably shield the radiation emitted from the loaded MPC, and that the HI-TRAC end plates (top lid and transfer lid for HI-TRAC 100 and HI-TRAC 125 and the top lid and pool lid for HI-TRAC 100D and 125D) remain attached. Furthermore, the HI-TRAC inner shell is demonstrated by analysis to not deform sufficiently to hinder retrieval of the MPC. The horizontal lift height limit is dependent on the characteristics of the impacting surface, which are specified in Table 2.2.9, and the number of fuel assemblies being stored (full or partially loaded MPC). For site-specific conditions including impact surfaces not encompassed by Table 2.2.9 or handling a partially loaded MPC, the licensee shall evaluate the site-specific conditions to ensure that the drop accident loads do not exceed 45 g's. The methodology used in this alternative analysis shall be commensurate with the methodology described in this FSAR and shall be reviewed by the Certificate Holder. The use of lifting devices designed in accordance with ANSI N14.6 having redundant drop protection features during horizontal lifting of the loaded HI-TRAC outside of the reactor facilities eliminate the need for a horizontal lift height limit.

The loaded HI-TRAC, when lifted in the vertical position outside of the Part 50 facility shall be lifted with devices designed in accordance with ANSI N14.6 and having redundant drop protection features unless a site-specific analysis has been performed to determine a lift height limit. For vertical lifts of HI-TRAC with suitably designed lift devices, a vertical drop is not a credible accident

analysis that is required to demonstrate compliance. Since the breakdown into specific analyses is most applicable to the structural evaluation, the identification of individual analyses with the applicable loads for each load combination is found in Chapter 3. Tables 3.1.3 through 3.1.5 define the particular evaluations of loadings that demonstrate compliance with the load combinations of Table 2.2.14.

For structural analysis purposes, Table 2.2.14 serves as an intermediate classification table between the definition of the loads (Table 2.2.13 and Section 2.2) and the detailed analysis combinations (Tables 3.1.3 through 3.1.5).

Finally, it should be noted that the load combinations identified in NUREG-1536 are considered as applicable to the HI-STORM 100 System. The majority of load combinations in NUREG-1536 are directed toward reinforced concrete structures. Those load combinations applicable to steel structures are directed toward frame structures. As stated in NUREG-1536, Page 3-35 of Table 3-1, "Table 3-1 does not apply to the analysis of confinement casks and other components designed in accordance with Section III of the ASME B&PV Code." Since the HI-STORM 100 System is a metal shell structure, with concrete primarily employed as shielding, the load combinations of NUREG-1536 are interpreted within the confines and intent of the ASME Code.

2.2.8 Allowable Stresses

The stress intensity limits for the MPC confinement boundary for the design condition and the service conditions are provided in Table 2.2.10. The MPC confinement boundary stress intensity limits are obtained from ASME Code, Section III, Subsection NB. The stress intensity limits for the MPC fuel basket are presented in Table 2.2.11 (governed by Subsection NG of Section III). The steel structure of the overpack and the HI-TRAC meet the stress limits of Subsection NF of ASME Code, Section III for plate and shell components. Limits for the Level D condition are obtained from Appendix F of ASME Code, Section III for the steel structure of the overpack. The ASME Code is not applicable to the HI-TRAC transfer cask for accident conditions, service level D conditions. The HI-TRAC transfer cask has been shown by analysis to not deform sufficiently to apply a load to the MPC, have any shell rupture, or have the top lid, pool lid, or transfer lid (as applicable) detach.

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

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

S_y: Minimum yield strength at temperature

S_u: Minimum ultimate strength at temperature

2.2.9 Requirements on Lifting and Special Lifting Devices

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2.2.9.1 Definitions: The lifting and handling systems used in Holtec's used fuel management program are made up of individual components or devices. These components can be further classified as either "*lifting devices*" or "*special lifting devices*." (See Glossary)

The term *special lifting device* refers to components to which ANSI N14.6 applies. As stated in ANSI N14.6 (both 1978 and 1993 versions), "This standard shall apply to *special lifting devices* that transmit the load from lifting attachments, which are structural parts of a container to the hook(s) of an overhead hoisting system." Examples of special lifting devices used with Holtec's systems include MPC lift cleats, lift brackets, and lift yokes

The term *lifting device* refers to components of a lifting and handling system that are not classified as *special lifting devices*. ANSI N14.6 is not applicable to these *lifting devices*. These include non-active structural components (components that bear the primary load but are not a constituent of a moving part, e.g., gear train, hydraulic cylinder) of the system. Examples of lifting devices used with Holtec's systems include: a vertical cask transporter's overhead beam, the structural members (viz. the main girder) of a gantry crane or a cask crane used to handle the MPC inside a part 50 structure.

The design of all lifting devices is governed by a Purchasing Specification prepared under the system designer's QA program which shall contain appropriate interpretations of the applicable codes and standards, required material properties, extreme environmental loadings (viz., earthquakes) and the like. The qualification for seismic and other applicable environmental loads is not required for transient states such as when the load is being emplaced and fastened or the lifting device is in motion.

2.2.9.2 Stress compliance criteria applicable to *Lifting Devices and Special Lifting Devices*:

The stress compliance criteria for *lifting devices* are taken from the code applicable to the specific component defined in the system designer's Purchasing Specification. For example, slings are required to meet the guidelines of ANSI B30.9 and overhead beams are required to meet the guidelines of an applicable consensus national standard selected by the designer, such as AISC, CMAA, or ASME Code (Subsection NF). Where a suitable consensus standard does not exist, the system designer is required to specify the necessary stress and strength requirements appropriate to the hardware.

The stress compliance criteria for *special lifting devices* are taken directly from ANSI N14.6, which requires safety factors of three against the yield strength and five times against ultimate strength under the dead load to be lifted.

2.2.9.3 Single Failure Proof Criteria

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

a *lifting device* to be considered single failure proof, the applicable code requirements should be doubled, or a redundant *lifting device* should be provided. The load drop protection feature incorporated in the vertical towers of a cask transporter is an example of redundant lifting part.

The horizontal transporters, referred to as Hauling transporter, Low Profile transporter, etc., are characterized by the absence of a lifting feature. Such ground supported equipment is considered single failure proof if the stresses developed under the design basis dead load are <50% of the allowable limit set down in the system designer's specification.

Likewise, for cask handling purposes, a plant's main crane can be treated as single failure proof if the structural factors of safety against the applicable code limit are a minimum of 2.

Similarly for a *special lifting device* to be considered *single failure proof*, the design safety factors in ANSI N14.6 should be doubled, or a redundant *special lifting device* should be provided.

Alternatively, the designer may perform a load drop analysis (permitted by both NUREG-0612 and ANSI N14.6). If the analyses support the conclusion that, after the physically admissible drop accident, the permissible dose rate from the cask does not exceed the plant's accident condition dose limit and the MPC meets the sub-criticality criterion of §72.124 then the increased safety factors are not required. In addition, for a drop scenario involving a loaded MPC, the confinement integrity of the MPC must remain intact and the MPC must remain retrievable subsequent to the drop event.

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TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM OVERPACK [1,2]

Primary Function	Component [3]	Safety Class [4]	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
					Section II		
Structural Integrity	Bolt Anchor Block	B	ASME Section III; Subsection NF Regulatory Guide 3.61 ~~ANSI N14.6~~	SA350-LF3, SA350-LF2, or SA203E	See Table 3.3.3	See Note 5	NA
Structural Integrity	Channel	B	ASME Section III; Subsection NF	SA516-70 (galvanized) or SA240-304	See Table 3.3.2 or Table 3.3.1	See Note 5	NA
Structural Integrity	Channel Mounts	B	ASME Section III; Subsection NF	SA36 or equivalent	Per ASME Section II	See Note 5	NA
Shielding	Pedestal Platform	B	Non-Code	SA36 or equivalent	NA	See Note 5	NA
Operations	Storage Marking Nameplate	NITS	Non-code	SA240-304	NA	NA	NA
Operations	Exit Vent Screen Sheet	NITS	Non-code	SA240-304	NA	NA	NA
Operations	Drain Pipe	NITS	Non-code	C/S or S/S	NA	See Note 5	NA
Operations	Exit & Inlet Screen Frame	NITS	Non-code	SA240-304	NA	NA	NA
Operations	Temperature Element &	C	Non-code	NA	NA	NA	NA

Notes:
1) There are no known residuals on finished component surfaces
2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
3) Component nomenclature taken from Bills of Material and drawings in Chapter 1. All components are "as applicable" based in the overpack drawing/BOM unless otherwise noted.
4) A, B, and C denote important to safety classifications as described in the Holtec QA Program. NITS stands for Not Important to Safety.
5) All exposed steel surfaces (except threaded holes) to be painted in accordance with Appendix 1.C, to the extent practical.
6) Welds will meet AWS D1.1 requirements for prequalified welds, except that welder qualification and weld procedures of ASME Code Section IX may be substituted.

TABLE 2.2.6

MATERIALS AND COMPONENTS OF THE HI-STORM 100 SYSTEM
HI-TRAC TRANSFER CASK [1,2]

Primary Function	Component [3]	Safety Class [4]	Codes/Standards (as applicable to component)	Material	Strength (ksi)	Special Surface Finish/Coating	Contact Matl. (if dissimilar)
Structural Integrity	Pool Lid Bolt	B	ASME Section III; Subsection NF	SA193-B7	See Table 3.3.4	NA	NA
Structural Integrity	Lifting Trunnion Block	B	ASME Section III; Subsection NF	SA350-LF3	See Table 3.3.3	See Note 5	NA
Structural Integrity	Lifting Trunnion	A	~~ANSI N14.6~~NUREG-0612 Regulatory Guide 3.61	SB637 (N07718) or SA564-630H1100 (For HI-TRAC125D only)	See Table 3.3.4	NA	NA
Structural Integrity	Pocket Trunnion (HI-TRAC 100 and HI-TRAC 125 only)	B	ASME Section III; Subsection NF NUREG-0612 Regulatory Guide 3.61~~ANSI N14.6~~	SA350-LF3	See Table 3.3.3	See Note 5	NA
Structural Integrity	Dowel Pins	B	ASME Section III; Subsection NF	SA564-630	See Table 3.3.4	NA	SA350-LF3
Structural Integrity	Water Jacket End Plate	B	ASME Section III; Subsection NF	SA516-70	See Table 3.3.2	See Note 5	NA

Notes:
1) There are no known residuals on finished component surfaces
2) All welding processes used in welding the components shall be qualified in accordance with the requirements of ASME Section IX. All welds shall be made using welders qualified in accordance with ASME Section IX. Weld material shall meet the requirements of ASME Section II and the applicable Subsection of ASME Section III.
3) Component nomenclature taken from Bill of Materials in Chapter 1.
4) A, B, and C denote important to safety classifications as described in the Holtec QA Program. NITS stands for Not Important to Safety.
5) All surfaces to be painted in accordance with Appendix 1.C, as applicable.

Table 2.2.7

HI-STORM 100 ASME BOILER AND PRESSURE VESSEL CODE APPLICABILITY

HI-STORM 100 Component	Material Procurement	Design	Fabrication	Inspection
Overpack steel structure	Section II, Section III, Subsection NF, NF-2000	Section III, Subsection NF, NF-3200	Section III, Subsection NF, NF-4000	Section III, Subsection NF, NF-5350, NF-5360 and Section V
Anchor Studs for HI-STORM 100A	Section II, Section III, Subsection NF, NF-2000*	Section III, Subsection NF, NF- 3300	NA	NA
MPC confinement boundary	Section II, Section III, Subsection NB, NB-2000	Section III, Subsection NB, NB-3200	Section III, Subsection NB, NB-4000	Section III, Subsection NB, NB-5000 and Section V
MPC fuel basket	Section II, Section III, Subsection NG, NG-2000; core support structures (NG-1121)	Section III, Subsection NG, NG-3300 and NG-3200; core support structures (NG-1121)	Section III, Subsection NG, NG-4000; core support structures (NG-1121)	Section III, Subsection NG, NG-5000 and Section V; core support structures (NG-1121)
HI-TRAC Trunnions	Section II, Section III, Subsection NF, NF-2000	NUREG-0612 and Regulatory Guide 3.61~~ANSI N14.6~~	Section III, Subsection NF, NF-4000	See Chapter 9
MPC basket supports (Angled Plates)	Section II, Section III, Subsection NG, NG-2000; internal structures (NG-1122)	Section III, Subsection NG, NG-3300 and NG-3200; internal structures (NG-1122)	Section III, Subsection NG, NG-4000; internal structures (NG-1122)	Section III, Subsection NG, NG-5000 and Section V; internal structures (NG-1122)
HI-TRAC steel structure	Section II, Section III, Subsection NF, NF-2000	Section III, Subsection NF, NF-3300	Section III, Subsection NF, NF-4000	Section III, Subsection NF, NF-5360 and Section V
Damaged fuel container	Section II, Section III, Subsection NG, NG-2000	Section III, Subsection NG, NG-3300 and NG-3200	Section III, Subsection NG, NG-4000	Section III, Subsection NG, NG-5000 and Section V
Overpack concrete	ACI 349 as specified by Appendix 1.D	ACI 349 and ACI 318.1-89(92) as specified by Appendix 1.D	ACI 349 as specified by Appendix 1.D	ACI 349 as specified by Appendix 1.D

* Except impact testing shall be determined based on service temperature and material type.

- Normal Condition: dead weight, handling of the Closure Lid, soil overburden pressure from subgrade, live load due to cask transporter movement, snow loads, and buoyancy effect of water saturation of surrounding subgrade and foundation. Most normal condition loadings occur at an ambient temperature denoted as the "normal storage condition temperature"; however, for calculations involving the Closure Lid, a higher temperature is assumed when the VVM carries a loaded MPC since the Closure Lid outlet ducts will be subject to heated air.

- Off-Normal Condition: elevated ambient temperature and partial blockage of air inlets.

- Extreme Environmental Phenomena and Accident Condition: handling accidents, fire, tornado, flood, earthquake, explosion, lightning, burial under debris, 100% blockage of air inlets, extreme environmental temperature, 100% fuel rod rupture, and an accident during construction in the vicinity of a loaded ISFSI.

The design basis magnitudes of the above loads, as applicable, are provided in Tables 2.I.1 and 2.I.5, and are discussed further in the following subsections. Applicable loads for an MPC contained in a VVM or for a HI-TRAC that services a VVM are identical to those already identified in the main body of Chapter 2 and, therefore, are not repeated or discussed within this supplement. However, recognizing that the support of an MPC in a VVM is different from the support provided in an above ground HI-STORM, the design basis dynamic analysis model includes the fuel assemblies, the fuel basket, and the enclosure vessel so that the loads described above are properly distributed within the VVM.

2.I.4 Normal Condition Operating Parameters and Loads

i. Dead Load

The HI-STORM 100U System must withstand the static loads due to the weight of each of its components. As the support provided by the subgrade and the VVM Interface Pad from lateral friction is apt to be negligible, the weight of the Closure Lid is assumed to bear on the Container Flange and the Container Shell; the load to the VVM Support Foundation is transferred through direct bearing action.

ii. Handling Loads

The only instance of a handling load occurs during emplacement or removal of the Closure Lid while the CEC contains a loaded MPC. To provide defense-in-depth, Closure Lid lifting attachments ~~shall~~ are conservatively designed to meet the design requirements of ANSI N14.6 [2.2.3].

Lift locations for the CEC and the Divider Shell are used for lifting only during construction, and possibly during maintenance and decommissioning of the VVM with no loaded MPC present; therefore, these lifting locations are not subject to the defense-in-depth measures of

basket) is the vehicle for performing the structural qualification. In addition to the primary stress limits, the local strain in the Confinement Boundary due to the impact between the MPC and the MPC guides under the Design Basis Earthquake requires evaluation.

Table 2.I.5 summarizes the above discussion in tabular form.

Load Case 05: Closure Lid Handling

The Closure Lid lifting attachments ~~shall~~ are conservatively designed to meet the strength limits of ANSI N14.6 for heavy load handling. The metal load bearing parts shall satisfy the requirements of Reg. Guide 3.61 for primary stresses near the lifting locations and shall satisfy ASME NF Level A limits away from the lifting locations.

Yield and ultimate strength values used in the stress compliance demonstration per ANSI N14.6 shall utilize confirmed material test data through either independent coupon testing or material suppliers' CMTR or COC, as appropriate.

Load Case 06: Design Basis Fire Event

The exposed portion of the VVM, namely the Closure Lid, will experience the heat input and temperature rise under the fire event. The balance of the VVM, because of its underground location, will be subject to only a secondary temperature increase.

It is required to demonstrate that the structural collapse of the Closure Lid cannot occur due to the reduction of its structural material's (low carbon steel) strength at the elevated temperatures from the fire.

Load Case 07: CEC Loading From Surrounding Subgrade

The CEC is subject to a lateral pressure from the soil in the non-seismic condition. This pressure is affected by the presence of a loaded cask transporter adjacent to the CEC. The CEC must be shown to provide adequate resistance to this loading.

This load case tends to ovalize the CEC; the maximum primary membrane plus bending stress is limited to the material yield strength under normal conditions of storage.

In evaluating the structural safety margins in Supplement 3.I for the load cases described above, design data for the interfacing SSCs presented in Table 2.I.2 is used as applicable.

2.I.10 Safety Protection Systems

The HI-STORM 100U System, featuring the VVM with the stored MPC, provides for confinement, criticality control, and heat removal for the stored spent nuclear fuel in the manner of the aboveground overpacks. The VVM provides better shielding and protection from environmental

TABLE 2.I.1 (continued)
LOADS, CRITERIA, APPLICABLE REFERENCE REGULATION/CODES AND STANDARDS FOR HI-STORM 100U VVM

Type	Criteria or Value and Reference Location in the FSAR	Basis, Regulation and Reference Code/Standard
Criticality:	N/A; Provided by MPC; Supplement 6.I	10CFR72.124 and 10CFR72.128(a)(2)
Radiation Protection/Shielding:		
Normal/Off-Normal	Provide capability to meet controlled area boundary dose limits under 10CFR72 for all normal and off-normal conditions; Supplement 5.I	10CFR72.104 and 10CFR72.212
	Ensure dose rates on and around the VVM during MPC transfer and lid installation operations are ALARA; Supplement 10.I	10CFR20
Accident or Conditions of Extreme Environmental Phenomena	Meet controlled area boundary dose limits in regulations for all accidents; Supplement 5.I	10CFR72.106
Design Bases:		
Spent Fuel Specification	Table 2.0.1; Section 2.I.1	10CFR72.236(a)
Normal Design Event Conditions:		
Ambient Outside Temperature:	-	-
Max. Yearly Average	80°F; Subsection 2.2.1.4	ANSI/ANS 57.9
Live Load†:		
Loaded HI-TRAC 125D and Mating Device	Table 3.I.1, Subsection 2.I.9	R.G. 3.61
Dry Loaded MPC	Table 3.I.1, Subsection 2.I.9	R.G. 3.61
Cask Transporter	Table 3.I.1, Subsection 2.I.9	-
Handling:	Subsection 2.I.4	-
VVM Closure Lid Lift Points	Subsection 3.I.4	NUREG-0612 ~~ANSI N14.6~~Regulatory Guide 3.61
Minimum Temperature During Closure Lid Handling Operations	0°F; Subsection 2.2.1.2	ANSI/ANS 57.9
Snow and Ice Load	100 lb/ft^2; Subsection 2.I.4	ASCE 7-88
Wet/Dry Loading	Dry; Supplement 1.I, 8.I	-
Storage Orientation	Vertical; Supplement 1.I	-

† Weights listed in Table 3.I.1 are bounding weights. Actual weights will be less, and will vary based on as-built dimensions of the components, fuel type, and the presence of fuel spacers and non-fuel hardware, as applicable.

TABLE 2.I.5
LOAD CASES AND ACCEPTANCE CRITERION APPLICABLE TO VVM COMPONENTS

Case I.D.	Bounding Loading	Affected Sub-Component	Applicable Data		Acceptance Criterion
			Magnitude of Loading (Ref. Table I.D.)	Value of Coincident Metal Temperature used (Deg. F)	
01	Condition with no MPC or Closure Lid installed; buoyancy from a water head equal to the distance between TOG and TOF.	• Temporary Cover • CEC Bottom Plate	Buoyant Force From CEC Displaced Volume < 8 psi	125 125	The minimum weight of the anti-buoyancy cover is 16,000lb. Maximum primary bending stress intensity in the CEC Bottom Plate must be below Level D limit.
02	Normal operation condition; dead load plus design basis explosion pressure	• Container Shell structure • Closure Lid	2.I.1; 3.I.1 2.I.1	125 350	Primary stresses do not exceed applicable Level A stress limits of ASME Subsection NF (or Level D limits with explosion)
03	Design basis missile	Closure Lid	2.I.1 and 2.2.5	350	Closure Lid does not collapse, is not dislodged from the cavity, and is not perforated by the missile.
04	Design basis earthquake	Container Shell	Figure 2.I.6	125	After the DBE event, MPC retrievability, subcriticality and confinement must not be compromised. Additional criteria for the CEC and its contents are defined in Table 2.I.6.
05	Closure lid handling	Lid Lift Lugs; all metal structure in Lid	1.15 x Closure Lid Weight (From Table 3.I.1)	125	~~ANSI N14.6~~NUREG-0612 and Regulatory Guide 3.61 limits based on yield or ultimate strength including magnified inertia loads. Meet Reg. Guide 3.61 and Level A limits as applicable. (see Section 2.I.9)

Same as Subsection 2.0.1. Generic operating procedures for the HI-STORM 100 System with MPC-68M are provided in Chapter 8 with certain limitations and clarifications provide in Supplement 8.III.

vii. Acceptance Tests and Maintenance

Same as Subsection 2.0.1. The acceptance criteria for the HI-STORM 100 System with MPC-68M are provided in Chapter 9 and Supplement 9.III.

vi. Decommissioning

The MPC is designed to be transportable in a HI-STAR overpack and is not required to be unloaded prior to shipment off-site. Decommissioning of the HI-STORM 100 System is addressed in Section 2.III.4.

2.III.1 SPENT FUEL TO BE STORED

Table 2.1.22 and the limitations/clarification in this supplement provide the limits for material to be stored in the MPC-68M. All BWR fuel assembly array/classes which are authorized for the MPC-68 are authorized in the MPC-68M except fuel assembly array/classes 6x6A, 6x6B, 6x6C, 7x7A, 8x8A, 10x10D, and 10x10E. Table 2.1.4 in Chapter 2 provides the acceptable fuel characteristics for the fuel array/class authorized for storage in the MPC-68M, however fuel with planar-average initial enrichments up to 4.8 wt.% U-235 are authorized in the MPC-68M. The maximum planar-average initial enrichments acceptable for loading in the MPC-68M, for each fuel assembly array/class given in Table 2.1.4, are provided in Table 2.III.2. Table 2.III.3 provides the description of two new fuel assembly array/classes which are added as acceptable contents to the MPC-68M only, 10x10F and 10x10G. No credit is taken for fuel burnup or integral poisons such as gadolinia for any fuel assembly array/class. The maximum allowable initial enrichment for fuel assemblies are consistent with the criticality analysis described in Supplement 6.III. ~~See Table 2.1.29 for assembly cooling time-dependent coefficients for 10x10F and 10x10G.~~

Fuel classified as damaged fuel assemblies or fuel debris will be loaded into damaged fuel containers (DFCs) for storage in the MPC-68M. The appropriate thermal and criticality analyses have been performed to account for damaged fuel and fuel debris and are described in Supplements 4.III and 6.III, respectively. The restrictions on the number and location of damaged fuel containers authorized for loading in the MPC-68M is the same as MPC-68 (see Section 1.III.2.3 and Figure 1.III.2). Non-fuel hardware is not applicable to all the BWR fuel classes/arrays.

The heat generation rate, axial burnup distribution, and all other bounding radiological, thermal, and criticality parameters specified for MPC-68 are used to ensure the performance of the HI-STORM SYSTEM with the MPC-68M.

2.III.2 MPC-68M DESIGN LOADINGS

Table 2.III.3: BWR FUEL ASSEMBLY CHARACTERISTICS FOR LOADING IN MPC-68M (Note 1)

Fuel Assembly Array and Class	**10x10F**	**10x10G**
Clad Material (Note 2)	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 192	≤ 188
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U) (Note 8, 9)	4.7 (Note 7)	4.~~6~~75 (Note 7)
Initial Rod Maximum Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 4)	96/84
Fuel Clad O.D. (in.)	≥ 0.4035	≥ 0.387
Fuel Clad I.D. (in.)	≤ 0.3570	≤ 0.340
Fuel Pellet Dia. (in.)	≤ 0.3500	≤ 0.334
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.512
Design Active Fuel Length (in.)	≤ 150	≤ 150
No. of Water Rods (Note 6)	2	5 (Note 5)
Water Rod Thickness (in.)	≥ 0.030	≥ 0.031
Channel Thickness (in.)	≤ 0.120	≤ 0.060

Table 2.III.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. See Table ~~1.0.1~~glossary for the definition of "ZR."

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

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

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

6. These rods may also be sealed at both ends and contain ZR material in lieu of water.

7. Fuel assemblies classified as damaged fuel assemblies are limited to 4.6 wt.% ^{235}U for the 10x10F and 10x10G arrays/class~~es and 4.0 wt.% 235U for the 10x10G array/class~~.

8. For MPC-68M loaded with both intact fuel assemblies and damaged fuel assemblies or fuel debris, the maximum planar average initial enrichment for the intact fuel assemblies is limited to the enrichment of the damaged assembly.

9. In accordance with the definition of UNDAMAGED FUEL ASSEMBLY, certain assemblies may be limited to up to 3.3 wt.% U-235. When loading these fuel assemblies, all other undamaged fuel assemblies in the MPC are limited to enrichments specified in this table and Table 2.III.2.

As discussed in Chapters 1 and 2, and Section 3.0, the principal shielding material utilized in the HI-STORM 100 Overpack is plain concrete. Plain concrete was selected for the HI-STORM 100 Overpack in lieu of reinforced concrete, because there is no structural imperative for incorporating tensile load bearing strength into the contained concrete. From a purely practical standpoint, the absence of rebars facilitate pouring and curing of concrete with minimal voids, which is an important consideration in light of its shielding function in the HI-STORM 100 Overpack. Plain concrete, however, acts essentially identical to reinforced concrete under compressive and bearing loads, even though ACI standards apply a penalty factor on the compressive and bearing strength of concrete in the absence of rebars (vide ACI 318.1).

Accordingly, the plain concrete in the HI-STORM 100 is considered as a structural material only to the extent that it may participate in supporting direct compressive loads. The allowable compression/bearing resistance is defined and quantified in the ACI 318.1-89(92) Building Code for Structural Plain Concrete.

In general, strength analysis of the HI-STORM 100 Overpack and its confined concrete is carried out only to demonstrate that the concrete is able to perform its radiation protection function and that retrievability of the MPC subsequent to any postulated accident condition of storage or handling is maintained.

A discrete ITS component in the HI-STORM 100 System is the HI-TRAC transfer cask. The HI-TRAC serves to provide a missile and radiation barrier during transport of the MPC from the fuel pool to the HI-STORM 100 Overpack. The HI-TRAC body is a double-walled steel cylinder that constitutes its structural system. Contained between the two steel shells is an intermediate lead cylinder. Attached to the exterior of the HI-TRAC body outer shell is a water jacket that acts as a radiation barrier. The HI-TRAC is not a pressure vessel since it contains a penetration in the HI-TRAC top lid that does not allow for a differential pressure to develop across the HI-TRAC wall. Nevertheless, in the interest of conservatism, structural steel components of the HI-TRAC are subject to the stress limits of the ASME Code, Section III, Subsection NF, Class 3.

Since both the HI-STORM 100 and HI-TRAC may serve as an MPC carrier, their lifting attachments are designed to meet the design safety factor requirements of NUREG-0612 [3.1.1] and ~~ANSI N14.6-1993 [3.1.2] for single-failure-proof lifting equipment~~Regulatory Guide 3.61 [1.0.2].

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

3.1.2 Design Criteria

Principal design criteria for normal, off-normal, and accident/environmental events are discussed in Section 2.2. In this section, the loads, load combinations, and allowable stresses used in the structural evaluation of the HI-STORM 100 System are presented in more detail.

Symbol	Description	Notes
	bending stress	structure. Occurs at structural discontinuities. Can be caused by pressure, mechanical loads, or differential thermal expansion.
F	Peak stress	Increment added to primary or secondary stress by a concentration (notch), or, certain thermal stresses that may cause fatigue but not distortion. This value is not used in the tables.

It is shown that there is no interference between component parts due to free thermal expansion. Therefore, P_e does not develop within any HI-STORM 100 component.

It is recognized that the planar temperature distribution in the fuel basket and the overpack under the maximum heat load condition is the highest at the cask center and drops monotonically, reaching its lowest value at the outside surface. Strictly speaking, the allowable stresses/stress intensities at any location in the basket, the enclosure vessel, or the overpack should be based on the coincident metal temperature under the specific operating condition. However, in the interest of conservatism, reference temperatures are established for each component, which are upper bounds on the metal temperature for each situational condition. Table 3.1.17 provides the reference temperatures for the fuel basket and the MPC canister utilizing Tables 3.1.6 through 3.1.16, and provides conservative numerical limits for the stresses and stress intensities for all loading cases. Reference temperatures for the MPC baseplate and the MPC lid are 400 degrees F and 550 degrees F, respectively, as specified in Table 2.2.3.

Finally, the lifting attachments or the interfacing lifting points ~~devices~~ in the HI-STORM 100 Overpack and HI-TRAC casks and the multi-purpose canisters must meet the requirements of NUREG-0612 and/or Regulatory Guide 3.61 as described in Subsection 3.4.3 and Tables 2.0.1, 2.0.2 and 2.0.3~~, collectively referred to as "trunnions", are subject to specific limits set forth by NUREG-0612: the primary stresses in a trunnion must be less than the smaller of 1/10 of the material ultimate strength and 1/6 of the material yield strength~~ under a normal handling condition (Load Case 01 in Table 3.1.5). The load combination D+H in Table 3.1.5 is equivalent to 1.15D. This is further explained in Subsection 3.4.3.

The region around the trunnions is part of the NF structure in HI-STORM 100 and HI-TRAC and NB pressure boundary in the MPC, and as such, must satisfy the applicable stress (or stress intensity) limits for the load combination. In addition to meeting the applicable Code limits, it is further required that the primary stress required to maintain equilibrium at the defined trunnion/mother structure interface must not exceed the material yield stress at three times the handling condition load (1.15D). This criterion, mandated by Regulatory Guide 3.61, Section 3.4.3, insures that a large safety factor exists on non-local section yielding at the trunnion/mother structure interface that would lead to unacceptable section displacement and rotation.

3.1.2.3 Brittle Fracture

The MPC canister and basket are constructed from a series of stainless steels termed Alloy X.

These stainless steel materials do not undergo a ductile-to-brittle transition in the minimum temperature range of the HI-STORM 100 System. Therefore, brittle fracture is not a concern for the MPC components. Such an assertion can-not be made a priori for the HI-STORM storage overpack and HI-TRAC transfer cask that contain ferritic steel parts. In general, the impact testing requirements for the HI-STORM overpack and the HI-TRAC transfer cask are a function of two parameters: the Lowest Service Temperature (LST) and the normal stress level. The significance of these two parameters, as they relate to impact testing of the overpack and the transfer cask, is discussed below.

In normal storage mode, the LST of the HI-STORM storage overpack structural members may reach -40°F in the limiting condition wherein the spent nuclear fuel (SNF) in the contained MPCs emits no (or negligible) heat and the ambient temperature is at -40°F (design minimum per Chapter 2: Principal Design Criteria). During the HI-STORM handling operations, the applicable lowest service temperature is 0°F (which is the threshold ambient temperature below which lifting and handling of the HI-STORM 100 Overpack or the HI-TRAC cask is not permitted by the Technical Specification). Therefore, two distinct LSTs are applicable to load bearing metal parts within the HI-STORM 100 Overpack and the HI-TRAC cask; namely,

LST = 0°F for the HI-STORM overpack during handling operations and for the HI-TRAC transfer cask during all normal operating conditions.

LST = -40°F for the HI-STORM overpack during all non-handling operations (i.e., normal storage mode).

Parts used to lift the overpack or the transfer cask, which include the anchor block in the HI-STORM 100 overpack, and the pocket trunnions, the lifting trunnions and the lifting trunnion block in HI-TRAC, will henceforth be referred to as "significant-to-handling" (STH) parts. The applicable design codes for these elements of the structure ~~is ANSI N14.6~~are NUREG-0612 and Regulatory Guide 3.61. All other parts of the overpack and the transfer cask will be referred to as "NF" components. It is important to ensure that all materials designated as "NF" or "STH" parts possess sufficient fracture toughness to preclude brittle fracture. For the STH parts, the necessary level of protection against brittle fracture is deemed to exist if the NDT (nil ductility transition) temperature of the part is at least 40° below the LST. Therefore, the required NDT temperature for all STH parts is -40°F.

It is well known that the NDT temperature of steel is a strong function of its composition, manufacturing process (viz., fine grain vs. coarse grain practice), thickness, and heat treatment. For example, according to Burgreen [3.1.3], increasing the carbon content in carbon steels from 0.1% to 0.8% leads to the change in NDT from -50°F to approximately 120°F. Likewise, lowering of the normalizing temperature in the ferritic steels from 1200°C to 900°C lowers the NDT from 10°C to -50°C [3.1.3]. It, therefore, follows that the fracture toughness of steels can be varied significantly within the confines of the ASME Code material specification set forth in Section II of the Code. For example, SA516 Gr. 70 (which is a principal NF material in the HI-STORM 100 Overpack) can have a maximum carbon content of up to 0.3% in plates up to four inches thick. Section II further permits normalizing or quenching followed by tempering to enhance fracture toughness. Manufacturing processes which have a profound effect on fracture

TABLE 3.2.1 (CONTINUED)
HI-STORM OVERPACK WEIGHT DATA

Item	Bounding Weight (lb)
HI-STORM 100S(243) Overpack†	
• Overpack top lid	25,500††
• Overpack w/ lid (empty)	270,000
• Overpack w/ fully loaded MPC-24	360,000
• Overpack w/ fully loaded MPC-32	360,000
• Overpack w/ fully loaded MPC-68/68F/68FF	360,000
• Overpack w/ fully loaded MPC-24E/EF	360,000
HI-STORM 100A Overpack†	Same as above
HI-STORM 100S Version B(218) Overpack (values in parentheses use high density concrete in overpack body)	
• Overpack top lid	29,000
• Overpack w/ lid (empty)	270,000 (305,000)
• Overpack w/ fully loaded MPC-24	360,000 (395,000)
• Overpack w/ fully loaded MPC-32	360,000 (395,000)
• Overpack w/ fully loaded MPC-68/68F/68FF	360,000 (395,000)
• Overpack w/ fully loaded MPC-24E/EF	360,000 (395,000)
HI-STORM 100S Version B(229) Overpack (values in parentheses use high density concrete in overpack body)	
• Overpack top lid	29,000
• Overpack w/ lid (empty)	270,000 (320,000)
• Overpack w/ fully loaded MPC-24	360,000 (410,000)
• Overpack w/ fully loaded MPC-32	360,000 (410,000)
• Overpack w/ fully loaded MPC-68/68F/68FF	360,000 (410,000)
• Overpack w/ fully loaded MPC-24E/EF	360,000 (410,000)

† The bounding weights for the HI-STORM 100S(232) and 100S(243) overpacks listed in the above table are based on a maximum concrete (dry) density of 160.8 pcf. For improved shielding effectiveness, higher density concrete (up to 200 pcf dry) can be poured in the radial cavity of each of the HI-STORM 100S overpacks. At 200 pcf, the bounding weights of an empty overpack and a fully loaded overpack increase to 320,000 lb and 410,000 lb, respectively. Higher density concrete cannot be used in the HI-STORM 100 or 100A overpacks.

†† Based on a maximum concrete (dry) density of 155 pcf. For improved shielding effectiveness, higher density concrete (up to 200 pcf dry) can be poured in the HI-STORM 100S lids. At 200 pcf, the bounding weight of the lid increases to 28,000 lb.

††† Based on the following maximum fuel assembly weights (as applicable):

1,680 lb per assembly (including non-fuel hardware) for PWR fuel that requires fuel spacers
1,720 lb per assembly (including non-fuel hardware) for PWR fuel that does not require fuel spacers
~~7~~830 lb per assembly (including channels and DFCs) for BWR fuel

In accordance with NRC Bulletin 96-04 [3.4.7], a review of the potential for chemical, galvanic, or other reactions among the materials of the HI-STORM 100 System, its contents and the operating environments, which may produce adverse reactions, has been performed. Table 3.4.2 provides a listing of the materials of fabrication for the HI-STORM 100 System and evaluates the performance of the material in the expected operating environments during short-term loading/unloading operations and long-term storage operations. As a result of this review, no operations were identified which could produce adverse reactions beyond those conditions already analyzed in this FSAR.

3.4.2 Positive Closure

There are no quick-connect/disconnect ports in the confinement boundary of the HI-STORM 100 System. The only access to the MPC is through the storage overpack lid, which weighs over 23,000 pounds (see Table 3.2.1). The lid is fastened to the storage overpack with large bolts. Inadvertent opening of the storage overpack is not feasible; opening a storage overpack requires mobilization of special tools and heavy-load lifting equipment.

3.4.3 Lifting Devices

As required by Reg. Guide 3.61, in this subsection, analyses for all lifting operations applicable to the deployment of a member of the HI-STORM 100 family are presented to demonstrate compliance with applicable codes and standards.

The HI-STORM 100 System has the following components and devices participating in lifting operations: lifting trunnions located at the top of the HI-TRAC transfer cask, lid lifting connections for the HI-STORM 100 lid and for other lids in the HI-TRAC transfer cask, connections for lifting and carrying a loaded HI-STORM 100 vertically, and lifting connections for the loaded MPC.

Analyses of HI-STORM 100 storage overpack and HI-TRAC transfer cask lifting devices are reported in this submittal. Analyses of MPC lifting operations are presented in the HI-STAR 100 FSAR (Docket Number 72-1008, Subsection 3.4.3) and are also applicable here.

The evaluation of the adequacy of the lifting devices entails careful consideration of the applied loading and associated stress limits. The load combination D+H, where H is the "handling load", is the generic case for all lifting adequacy assessments. The term D denotes the dead load. Quite obviously, D must be taken as the bounding value of the dead load of the component being lifted. In all lifting analyses considered in this document, the handling load H is assumed to be 0.15D. In other words, the inertia amplifier during the lifting operation is assumed to be equal to 0.15g. This value is consistent with the guidelines of the Crane Manufacturer's Association of America (CMAA), Specification No. 70, 1988, Section 3.3, which stipulates a dynamic factor equal to 0.15 for slowly executed lifts. Thus, the "apparent dead load" of the component for stress analysis purposes is $D^* = 1.15D$. Unless otherwise stated, all lifting analyses in this report use the "apparent dead load", D^*, as the lifted load.

Analysis methodology to evaluate the adequacy of the lifting device may be analytical or

numerical. For the analysis of the trunnion, an accepted conservative technique for computing the bending stress is to assume that the lifting force is applied at the tip of the trunnion "cantilever" and that the stress state is fully developed at the base of the cantilever. This conservative technique, recommended in NUREG-1536, is applied to all trunnion analyses presented in this SAR and has also been applied to the trunnions analyzed in the HI-STAR 100 FSAR.

In general, the stress analysis to establish safety pursuant to NUREG-0612, Regulatory Guide 3.61, and the ASME Code, requires evaluation of three discrete zones which may be referred to as (i) the trunnion, (ii) the trunnion/component interface, hereinafter referred to as Region A, and (iii) the rest of the component, specifically the stressed metal zone adjacent to Region A, herein referred to as Region B. During this discussion, the term "trunnion" applies to any device used for lifting (i.e., trunnions, lift bolts, etc.)

Stress limits germane to each of the above three areas are discussed below:

i. Trunnion: NUREG-0612 and Reg. Guide 3.6.1 requires that under the "apparent dead load", D^*, the maximum primary stress in the trunnion be less than 10% of the trunnion material ultimate strength and less than 1/36rdth of the trunnion material yield strength. Because of the materials of construction selected for trunnions in all HI-STORM 100 System components, the ultimate strength-based limit is more restrictive in every case. Therefore, all trunnion safety factors reported in this document pertain to the ultimate strength-based limit.

ii. Region A: Trunnion/Component Interface: Stresses in Region A must meet ASME Code Level A limits under applied load D*. Additionally, Regulatory Guide 3.61 requires that the primary stress under 3D*, associated with the cross-section, be less than the yield strength of the applicable material. In cases involving section bending, the developed section moment may be compared against the plastic moment at yield. The circumferential extent of the characteristic cross-section at the trunnion/component interface is calculated based on definitions from ASME Section III, Subsection NB and is defined in terms of the shell thickness and radius of curvature at the connection to the trunnion block. By virtue of the construction geometry, only the mean shell stress is categorized as "primary" for this evaluation.

iii. Region B: Typically, the stresses in the component in the vicinity of the trunnion/component interface are higher than elsewhere. However, exceptional situations exist. For example, when lifting a loaded MPC, the MPC baseplate, which supports the entire weight of the fuel and the fuel basket, is a candidate location for high stress even though it is far removed from the lifting location (which is located in the top lid).

Even though the baseplate in the MPC would normally belong to the Region B category, for conservatism it was considered as Region A in the HI-STAR 100 SAR. The pool lid and the transfer lid of the HI-TRAC transfer cask also fall into

this dual category. In general, however, all locations of high stress in the component under D^* must also be checked for compliance with ASME Code Level A stress limits.

The acceptance criteria for lifting operations summarized above are consistent with those used in the HI-STORM FW FSAR, which has been approved by the NRC.

Unless explicitly stated otherwise, all analyses of lifting operations presented in this report follow the load definition and allowable stress provisions of the foregoing. Consistent with the practice adopted throughout this chapter, results are presented in dimensionless form, as safety factors, defined as

$$\text{Safety Factor, } \beta = \frac{\text{Allowable Stress in the Region Considered}}{\text{Computed Maximum Stress in the Region}}$$

The safety factor, defined in the manner of the above, is the added margin over what is mandated by the applicable code (NUREG-0612 or Regulatory Guide 3.61).

In the following subsections, we briefly describe each of the lifting analyses performed to demonstrate compliance with regulations. Summary results are presented for each of the analyses.

It is recognized that stresses in Region A are subject to two distinct criteria, namely Level A stress limits under D* and yield strength at 3D*. We will identify the applicable criteria in the summary tables, under the column heading "Item", using the "3D*" identifier.

All of the lifting analyses reported on in this Subsection are designated as Load Case 01 in Table 3.1.5.

3.4.3.1 125 Ton HI-TRAC Lifting Analysis - Trunnions

The lifting device in the HI-TRAC 125 cask is presented in Holtec Drawing 1880 (Section 1.5 herein). The two lifting trunnions for HI-TRAC are spaced at 180 degrees. The trunnions are designed for a two-point lift in accordance with the aforementioned NUREG-0612 criteria. Figure 3.4.21 shows the overall lifting configuration. The lifting analysis demonstrates that the stresses in the trunnions, computed using the conservative methodology described previously, comply with NUREG-0612 provisions.

Specifically, the following results are obtained:

HI-TRAC 125 Lifting Trunnions[†]		
	Value (ksi)	Safety Factor
Bending stress	16.09	1.13
Shear stress	7.26	1.50
[†] The lifted load is 245,800 lb.(a value that bounds the actual lifted weight from the pool after the lift yoke weight is eliminated per Table 3.2.4).		

Note that the safety factor presented in the previous table represents the additional margin beyond the mandated limit of 36 on yield strength and 10 on tensile strength.

Similar calculations have been performed for the HI-TRAC 125D cask, which differs from the HI-TRAC 125 with respect to the material options for the lifting trunnions. The lifting trunnions for the HI-TRAC 125 are fabricated from SB637-N07718; the lifting trunnions for the HI-TRAC 125D can be fabricated from either SB637-N07718 or SA564-630. The bounding results for the HI-TRAC 125D are:

HI-TRAC 125D Lifting Trunnions[†]		
	Value (ksi)	Safety Factor
Bending stress	13.57	1.03
Shear stress	7.26	1.16
[†] The lifted load is 245,800 lb.(a value that bounds the actual lifted weight from the pool after the lift yoke weight is eliminated per Table 3.2.4).		

3.4.3.2 125 Ton HI-TRAC Lifting - Trunnion Lifting Block Welds, Bearing, and Thread Shear Stress (Region A)

As part of the Region A evaluation, the weld group connecting the lifting trunnion block to the inner and outer shells, and to the HI-TRAC top flange, is analyzed. Conservative analyses are also performed to determine safety factors for bearing stress and for thread shear stress at the interface between the trunnion and the trunnion block. The following results are obtained for the HI-TRAC 125 and 125D transfer casks:

The following table summarizes the minimum safety factors from these analyses. As stated earlier, safety factors tabulated below represent margins that are over and beyond those implied by the loading magnification mandated in NUREG 0612 or Regulatory Guide 3.61, as appropriate.

Summary of MPC Lifting Analyses			
Item	**Thread Engagement Safety Factor (NUREG-0612/Reg. Guide 3.61)**	**Region A Safety Factor (Note 1)**	**Region B Safety Factor(Note 1)**
MPC	1.67~~1.013 (Note 2)~~	1.54	1.08

Notes:

1. Safety factor is for MPC baseplate.
2. ~~The Safety Factor is calculated at 475°F based on a minimum yield strength of 33 ksi at room temperature for MPC Lids.~~

When dual lids are used on the MPC, the outer lid transfers the entire lifted load to the peripheral weld. The maximum bending stress in the outer lid from the lifted load can be conservatively computed by strength of materials theory using the solution for a simply supported circular plate under a central concentrated load equal to 115% of the bounding MPC load. The calculation and result are presented below using tabular results from Timoshenko, Strength of Materials, Vol. II, 3rd Edition.

P = 90,000 lb. x 1.15
Outer Diameter a = 67.375"
Effective Central Diameter where load is applied b= 13.675" (conservative assumption)
a/b = 5
Lid thickness = 4.75" (Dual lids)
From the reference, k=1.745 and the maximum bending stress under the amplified lifted load is

$\sigma = kP/h^2 = 8005$ psi

Table 3.4.7 provides results for the stress in the lid under normal condition internal pressure. For the case with dual lids, the stress must be doubled. From the table, the pressure stress is

S = 2 x 1,633 psi

Therefore, the combined bending stress at the center of the dual lid is 11,271. Using the allowable strength from Table 3.4.7, the safety factor is

SF = 25,450 psi/11,271 psi = 2.258

3.4.4.3.3.5 HI-TRAC Top Lid Separation (Load Case 02.b in Table 3.1.5)

The potential of top lid separation under a 45g deceleration side drop event requires evaluation. It is concluded by analysis that the connection provides acceptable protection against top lid separation. It is also shown that the bolts and the lid contain the MPC within the HI-TRAC cavity during and after a drop event. The results from the HI-TRAC 125 bound the corresponding results from the HI-TRAC 100 because the top lid bolts are identical in the two units and the HI-TRAC 125 top lid weighs more. The analysis also bounds the HI-TRAC 125D and the HI-TRAC 100D because the postulated side drop of the HI-TRAC 125, during which the transfer lid impacts the target surface, produces a larger interface load between the MPC and the top lid of the HI-TRAC than the nearly horizontal drop of the HI-TRAC 125D and the HI-TRAC 100D. The table below provides the results of the bounding analysis.

HI-TRAC Top Lid Separation Analysis			
Item	**Value**	**Capacity**	**Safety Factor= Capacity/Value**
Attachment Shear Force (lb.)	123,750	957,619	7.738
Tensile Force in Stud (lb.)	132,000	1,117,222	8.464
Bending Stress in Lid (ksi)	35.56	58.7	1.65
Shear Load per unit Circumferential Length in Lid (lb./in)	533.5	29,400	55.10

3.4.4.4 Comparison with Allowable Stresses

Consistent with the formatting guidelines of Reg. Guide 3.61, calculated stresses and stress intensities from the finite element and other analyses are compared with the allowable stresses and stress intensities defined in Subsection 3.1.2.2 per the applicable sections of [3.4.2] and [3.4.4] for defined normal and off-normal events and [3.4.3] for accident events (Appendix F).

3.4.4.4.1 MPC

In Amendment #5 to the HI-STORM CoC, the weight limits for fuel assemblies to be stored in the MPCs were increased from 1,680 lbs to 1,720 lbs per assembly for PWR fuel. In order to account for this increase in fuel weight, the results of the MPC stress analysis under lateral loading, which is described in Subsection 3.4.4.3.1.1, are uniformly scaled based on the percentage weight increase. Specifically, the results for the PWR MPCs are scaled by a factor of 1.024 (=1720/1680). This approach is acceptable because (i) the finite element analysis results are based on linear elastic material properties and (ii) the percentage increases in total weight, considering the stored fuel, fuel basket, and MPC shell, are less than the factors above.

Finally, since the stresses associated with closing the support clearance gaps between the fuel basket and the MPC shell and between the MPC shell and the overpack are secondary stress components, as explained in Subsection 3.4.4.3.1.1, the use of a linear scale factor is an appropriate means of computing the primary stresses in the fuel basket and MPC shell.

Table 3.4.6 provides summary data extracted from the numerical analysis results for the fuel basket, enclosure vessel, and fuel basket supports after scaling to adjust for the increased fuel assembly weights. The results presented in Table 3.4.6 are based on the design basis deceleration and do not include any dynamic amplification due to internal elasticity of the structure (i.e., local inertia effects). Calculations suggest that a uniform conservative dynamic amplifier for the fuel basket would be 1.08 independent of the duration of impact. If we recognize that the tip-over event for HI-STORM 100 is a long duration event, then a dynamic amplifier of 1.04 is appropriate. The summary data provided in Table 3.4.3 and 3.4.4 gives the lowest safety factor computed for the fuel basket and for the MPC, respectively. Safety factors reported for the MPC shell in Table 3.4.4 are based on allowable strengths at 500 deg. F. Modification of the fuel basket safety factor for dynamic amplification leaves considerable margin. Factors of safety greater than 1 indicate that calculated results are less than the allowable strengths.

A perusal of the results in Tables 3.4.3 and 3.4.4 under different load combinations for the fuel basket and the enclosure vessel reveals that all factors of safety are above 1.0 even if we use the most conservative value for dynamic amplification factor. The relatively modest factor of safety in the fuel basket under side drop events (Load Case F3.b and F3.c) in Table 3.4.3 warrants further explanation since a very conservative finite element model of the structure has been utilized in the analysis.

The legacy 2-D finite element model dating back to the 1990s employed to simulate the MPC and the fuel basket, and other simplifications such as modeling the overpack as a rigid body and fuel as a pressure loading in some loading simulations used in this FSAR, lead to an understatement and associated uncertainty~~)~~ in the computed safety margins which can be can be alleviated by a 3-D analysis (under the §72.48 process). The use of a 3-D analysis ~~which~~ has been utilized on the ANSYS and LS-DYNA platform (as appropriate) and approved by the NRC in the HI-STORM FW docket.

It should be noted that the change of the BWR fuel weight ~~from 780 to 830 lb~~in Table 2.1.22 applies only to the specified DFC bearing locations but does not require the design basis gross weight of the MPC used in the structural analyses to be changed. Therefore, other safety analyses such as non-mechanistic tip-over and lifting and handling appurtenances remain unaffected and do not need to be revisited.

The wall thickness of the storage cells, which is by far the most significant variable in a fuel basket's structural strength, is significantly greater in the MPCs than in comparable fuel baskets licensed in the past. ~~For example, the cell wall thickness in the TN-32 basket (Docket No. 72-1021, M-56), is 0.1 inch and that in the NAC-STC basket (Docket No. 71-7235) is 0.048 inch. In contrast, the cell wall thickness in the MPC-68 is 0.25 inch.~~ In spite of their relatively high flexural rigidities, computed margins in the fuel baskets are rather modest. This is because of some assumptions in the analysis that lead to an overstatement of the state of stress in the fuel

TABLE 3.4.6
MINIMUM SAFETY FACTORS FOR MPC COMPONENTS DURING TIP-OVER 45g DECELERATIONS

Component - Stress Result	MPC-24		MPC-68	
	0 Degrees	**45 Degrees**	**0 Degrees**	**45 Degrees**
Fuel Basket - Primary Membrane (P_m)	3.38 (1134)	4.72 (396)	2.89 (1603)	4.18 (1603)
Fuel Basket - Local Membrane Plus Primary Bending (P_L+P_b)	1.29 (1065)	1.30 (577)	1.97~~2.09~~ (1590)	1.35~~1.38~~ (1459~~774~~)
Enclosure Vessel - Primary Membrane (P_m)	6.39*.967 (1354)	6.46*.967 (1370)	6.34~~6.29*.967~~ (2393)	6.64~~6.58*.967~~ (2377)
Enclosure Vessel - Local Membrane Plus Primary Bending (P_L+P_b)	2.46*.967 (1278)	2.92*.967 (1247)	1.02~~1.05*.967~~ (1925)	1.45~~1.50*.967~~ (1925)
Basket Supports – Primary Membrane (P_m)	N/A	N/A	6.61~~6.86~~ (1710)	8.61~~8.98~~ (1699)
Basket Supports - Local Membrane Plus Primary Bending (P_L+P_b)	N/A	N/A	1.09~~1.13~~ (1715)	1.43~~1.50~~ (1704)

Notes:

1. Corresponding ANSYS element number shown in parentheses.
2. Multiplier of 0.967 reflects increase in Enclosure Vessel Design Temperature from 450 deg. F to 500 deg. F (Table 2.2.3); tabulated results for MPC-68 are based on higher temperature (500 deg. F) for Enclosure Vessel.
3. Safety factors for the MPC-24 ~~and MPC-68~~ have been reduced (divided by factors of 1.024 ~~and 1.043, respectively~~) to adjust for the fuel assembly weight increase (see Subsection 3.4.4.4.1)

Design Criteria for the Impressed Current Cathodic Protection System
a. The cathodic protection system shall be capable of maintaining the CEC at a minimum (cathodic) potential as required by NACE Standard RP0285-2002 [3.I.21]. b. The ICCPS shall include provisions to infer its proper operation and effectiveness on a periodic basis. c. The system shall be designed to mitigate corrosion of the CEC for its design life. d. The cathodic protection system design, installation, operation, testing, and maintenance shall follow the applicable guidelines of: - 49CFR195 Subpart H "Corrosion Control", Oct. 1, 2004 edition [3.I.13] - NACE Standard RP0285-2002 "Corrosion Control of Underground Storage Tank Systems by Cathodic Protection" [3.I.21]

The following standards and/or publications may also be utilized for additional guidance in the design, installation, operation, testing, and maintenance of the ICCPS as needed (in case of conflict, the guidelines of item d above shall prevail):

- API RP1632, "Cathodic Protection of Underground Petroleum Storage Tanks and Piping Systems" [3.I.22]
- NACE RP0169-96, "Control of External Corrosion on Underground or Submerged Piping Systems" [3.I.23]
- 49CFR192 Subpart I, "Requirements for Corrosion Control", Oct. 1, 2004 edition [3.I.24]
- Other standards or publications referenced by any of the above three standards and publications.

Records of system operating data necessary to adequately track the operable status of the ICCPS shall be maintained in accordance with the user's quality assurance program.

Finally, the surface preservative used to coat the CEC must meet the requirements described in (i) above but must also be compatible with cathodic protection and resistant to the alkaline conditions created by cathodic protection and/or concrete encasement. Organic coatings, such as the Keeler & Long coating selected for (i) above, are inherently compatible with both cathodic protection [3.I.11] and concrete [3.I.10].

3.I.4.2 Positive Closure

There are no quick-connect/disconnect ports in the confinement boundary of the HI-STORM 100U system. Because the only access to the MPC is through the VVM Closure Lid, which weighs well over 10 tons, inadvertent opening of the VVM cavity is not feasible.

3.I.4.3 Lifting Devices

As required by Reg. Guide 3.61, lifting operations applicable to the VVM lid are analyzed. Because of the nature of the HI-STORM 100U system, lid placement or removal may occur with a loaded MPC inside the VVM cavity; these are the sole operations requiring analysis in accordance with Reg. Guide 3.61 and are examined in this supplement.

As discussed in Subsection 3.4.3, the lifting component itself (the four lift lugs) must meet the primary stress limits prescribed by NUREG-0612 and Regulatory Guide 3.61~~ANSI N14.6-1993~~; the welds in the load path, near the lifting holes, are required to meet the condition that stresses remain below yield under three times the lifted load (per Reg. Guide 3.61). Further, for additional conservatism, away from the lifting location, the ASME Code limit for the Level A service condition applies.

The lifting analysis results summarized below include a 15% inertia amplifier.

HI-STORM 100U VVM Closure Lid Lifting Analysis (Load Case 05 in Table 2.I.5)

The four lifting lugs are conservatively analyzed to ANSI N14.6 stress limits using simple strength of materials calculations. Each of four lugs is considered as a cantilever beam attached to the lid and carries 25% of the lid weight. The bending moment and shear force at the root of the cantilever (where it is attached to the lid) is computed and the maximum stress is conservatively compared with the minimum of the yield strength/6 or the ultimate strength/10. As required, increasing the lid weight by 15% includes inertia effects. Using the calculated bending moment and shear force at the root of the lug, the structural evaluation of the weld attaching the lug to the lid is performed and compared with the requirements of Regulatory Guide 3.61. The results from these two calculations demonstrate that the required safety factors are substantially greater than 1.0 (exceeding the requirements of ~~ANSI N14.6~~NUREG-0612 and Reg. Guide 3.61, respectively). The details of the calculations are presented in the calculation package supporting this submittal [3.I.27]. Lifting slings that attach to the lugs shall be sized to meet the safety factors set forth in ANSI B30.3.

To evaluate the global state of stress in the lid body, a finite element model of the lid, which includes contact interfaces between steel and concrete, is constructed to evaluate the state of stress under lifting conditions. Figure 3.I.1 shows the constructed ANSYS finite element model. The lifted scenario is simulated by fixing the four lifting locations at the lift lug sling attachment location, and applying an appropriate weight density to match the lifted weight. The results are evaluated for satisfaction of normal condition (ASME Level A) limits at the appropriate locations.

The table below summarizes key results obtained from the lifting analyses for the HI-STORM 100U VVM Closure Lid for a bounding set of input design loads.

HI-STORM 100U VVM Lid Lifting Analyses (Load Case 05 in Table 2.I.5)			
Item	Calculated Value	Allowable	Safety Factor
Bending of Lift Lugs (kip)(ANSI N14.6)	4.000	5.275	1.32 (see Note 1)
Shear in Lift Lugs (kip)(ANSI N14.6)	1.609	3.165	1.97 (see Note 1)
Load in Welds Near Lifting Lugs (kip) (Reg. Guide 3.61)	5.657	6.33	1.12 (see Note 2)
Primary Stress in Lid (ksi)(ASME Level A Limit)	< 10	26.25	> 2.63
Note 1: Computed safety factors represent the margin over that required by ~~ANSI N14.6-1993~~NUREG-0612 (0.1 x ultimate load). Note 2: Computed safety factor is based on 60% of yield strength for base metal and represents margin over limit set by Reg. Guide 3.61.			

It is concluded that all structural integrity requirements are met during a lift of the HI-STORM 100U VVM Closure Lid. All factors of safety, using applicable criteria from the ASME Code Section III, Subsection NF for Class 3 plate and shell supports, from USNRC Regulatory Guide 3.61, and from ~~ANSI N14.6~~NUREG-0612, are greater than 1.0.

3.I.4.4 Heat

i. Summary of Pressures and Temperatures

Tables 2.I.1 and 2.I.2 present applicable design inputs for the HI-STORM 100U VVM. No new inputs are required for the HI-TRAC and the MPC.

ii. Differential Thermal Expansion

All clearances between the MPC and the HI-STORM 100U VVM are equal to or larger than the corresponding clearances in the aboveground HI-STORM 100 systems (see Section 4.4). Therefore, no interferences between the MPC and the VVM will occur due to thermal expansion of the loaded MPC. The Divider Shell is insulated on one surface and is exposed to heated air on the other shell surface. Therefore an analysis to demonstrate that free axial thermal expansion of the Divider Shell will not close the initial gap between the top end of the Divider Shell and the base of the Closure Lid is provided. The Divider Shell is considered as a heated member, subject to an average temperature increase over its entire length. The actual axial absolute temperature profile can be integrated over the length of the Divider Shell to define the average absolute temperature. Once the average absolute temperature is known, the free thermal growth is computed and compared with the provided gap between the Divider Shell and the Closure Lid.

The average temperature rise above ambient is bounded by DT (ambient is 80^{o}F per Table 2.I.1, and average metal temperature over the length of the Divider Shell is from Table 4.I.3, footnote):

During a non-mechanistic tip-over event, the fuel assemblies exert a lateral force on the fuel basket panels as the overpack impacts the ground and decelerates. The lateral force causes the fuel basket panels to deflect potentially affecting the spacing between stored fuel assemblies. To maintain the fuel in a subcritical configuration, a deflection limit for the fuel basket panels is set in Subsection 2.III.0.1, which is supported by the criticality safety analysis in Supplement 6.III. Here a finite element analysis is performed using ANSYS to demonstrate that the maximum lateral deflection in the fuel basket panels under a bounding deceleration of 60~~70~~g is less than the limit specified in Section 2.III.0.1. The 60~~70~~g input deceleration is bounding because it exceeds the design basis deceleration limit of 45g for the non-mechanistic tip over of the HI-STORM storage overpack (see Subsection 3.III.4.10), ~~as well as~~and it matches the design basis lateral deceleration limit of 60g for the HI-STAR transport cask [1.1.3] for future considerations. The analysis methodology presented in this subsection is identical to the methodology used in [2.III.6.2] to qualify the F-37 fuel basket.

As shown in Figure 3.III.1, a representative slice of the MPC-68M fuel basket, consisting of a smaller end section and a full section, is modeled in detail including the contained fuel assemblies and supporting basket shims. The fuel basket panels are modeled with SOLSH190 solid shell elements. The basket shims and each fuel assembly are modeled with SOLID45 solid elements. The mass density assigned to the fuel assemblies corresponds to the maximum BWR fuel assembly weight per Table 2.1.22, except at the 16 cell locations along the basket perimeter where Damaged Fuel Containers are permitted. At these 16 locations, the mass density corresponds to the maximum weight of a BWR fuel assembly plus DFC per Table 2.1.22. Standard contact pairs using CONTA173/TARGE170 elements are defined at the interfaces of fuel assembly/basket panel, shim/basket panel, and between stacked basket panels including all the intersecting slot locations. The fuel basket material model is implemented with true stress-true strain multi-linear isotropic hardening plasticity model. An elastic material model is used for the basket shims since no plastic deformation is expected. To accommodate large plastic deformation in the fuel basket panels, sufficiently small element sizes (< 0.40 in) are used and 9 integration points through the thickness are specified. A sensitivity study was performed in [2.III.6.2] to confirm that the panel stresses and displacements obtained using solid shell elements are converged and comparable to those obtained using 5 solid elements through the thickness of the panel.

The 60~~70~~g deceleration is applied to the model with the basket in the so-called 0° orientation (see Figure 3.III.5). This orientation is chosen for analysis because it maximizes the lateral load on a single basket panel, which in turn maximizes the lateral deflection of the panel. In the 0° orientation, the amplified weight of each stored fuel assembly (during the 60~~70~~g impact event) bears entirely on one basket panel. Conversely, in the 45° orientation, the amplified weight of each stored fuel assembly is equally supported by two basket panels. The difference in loading between these two basket orientations is pictorially shown in Figure 3.III.5, where "m" denotes the fuel assembly mass, "a" denotes the maximum lateral deceleration, and "d" denotes the enveloping size of the fuel assembly. For comparison purposes, the pressure loads on the basket panels are defined as "p" and "q", respectively, for the 0° and 45° orientations. From the figure, the pressure load p that develops in the 0° orientation is 41% greater than the pressure load q that develops in the 45° orientation. Hence, the lateral deflection of a basket panel is much greater for

the 0° orientation (which is why it is chosen for detailed analysis). It is also noted that the 90° corners where the basket panels intersect do not provide any additional moment resistance because of the slotted joint construction (see Figure 1.III.1); therefore, the 45° orientation (or any other orientation between 0° and 45°) does not give rise to any prying loads at the cell corners. Finally, to ensure that the analysis for the 0° orientation is conservative and bounds all other basket orientations, the analysis is performed based on a lateral impact deceleration of 60~~70~~g even though, according to the results presented in Section 3.III.4.10, the maximum impact deceleration due to the non-mechanistic tip over event (measured at the top of the overpack lid) is less than 45g.

The stress and strain distributions in the fuel basket panels at 60~~70~~g are shown in Figures 3.III.2 and 3.III.3, respectively. These figures show that the state of stress in the fuel basket panels is primarily elastic. The fuel basket displacements are plotted in Figure 3.III.4. Table 3.III.4 compares the maximum lateral displacement in a fuel basket panel (relative to its end supports) with the deflection limit specified in Subsection 2.III.0.1.

Per the licensing drawing, the nominal width of fuel basket panels in the vertical direction may be increased or decreased provided that the length of the panel slots is increased or decreased proportionally. This means that the fixed-height fuel basket may be assembled using more (or fewer) panels than the number depicted on the licensing drawing. The results of the ANSYS static analysis for the fuel basket presented herein are valid for any panel width since (a) the lateral load on the fuel basket per unit (vertical) length remains the same and (b) the length of the slots measured as a percentage of the panel width remains the same.

Finally, to evaluate the potential for crack propagation and growth for the MPC-68M fuel basket under the non-mechanistic tipover event, a crack propagation analysis is carried out for the MPC-68M fuel basket using the same methodology utilized in Attachment D of [1.III.A.3] to evaluate the HI-STAR 180 F-37 fuel basket in support of the HI-STAR 180 Transport Package [2.III.6.2].

The crack propagation analysis is informed by the results from the ANSYS finite element analysis of the MPC-68M fuel basket under a bounding load of 60~~70~~-g, which is described above. In particular, the stress distribution in the Metamic-HT basket panels, as determined by ANSYS, is shown in Figure 3.III.2. The maximum stress occurs at one of the basket notches, which are conservatively modeled as sharp (90 degree) corners in the finite element model. This peak stress is used as input to the following crack propagation analysis.

Per [1.III.A.3] the critical stress intensity factor of Metamic-HT panels is estimated to be

$$K_{IC} = 30 ksi\sqrt{in}$$

based on Charpy V-notch absorbed energy (CVE) correlations for steels. The estimated value is consistent with the range for aluminum alloys, which is 20 to 50 $MPa\sqrt{m}$ or 18.2 to 45 $ksi\sqrt{in}$ per Table 3 of [3.III.4]. Next the minimum crack size, a_{min}, for crack propagation to occur is

calculated below using the formula for a through-thickness edge crack given in [3.1.5]. Although the formula is derived for a straight-edge specimen, the use of the peak stress, σ_{max}, at a notch in the fuel basket panel (instead of the average stress in the panel as required by the formula) essentially compensates for the geometric difference between the basket panel and the specimen. Moreover, the maximum size of a pre-existing crack (1/16") in the fuel basket panel is less than 1/6th of the basket panel thickness (0.40"). Thus, the assumption of a through-thickness edge crack is very conservative. The result is

$$a_{min} = \frac{\left(\frac{K_{IC}}{1.12\sigma_{max}}\right)^2}{\pi} = \frac{\left[\frac{30ksi\sqrt{in}}{1.12(17.426ksi)}\right]^2}{\pi} = 0.752in$$

and the safety factor against crack propagation (based on a 1/16" minimum detectable flaw size) is

$$SF = \frac{a_{min}}{a_{det}} = \frac{0.752in}{0.0625in} = 12.0$$

The calculated minimum crack size is ~~more than 11~~ 12 times greater than the maximum possible pre-existing crack size in the fuel basket (based on 100% surface inspection of each panel). The large safety factor ensures that crack propagation in the MPC-68M fuel basket will not occur due to the non-mechanistic tipover event.

3.III.4.4.3.2 Elastic Stability and Yielding of the MPC-68M Fuel Basket under Compression Loads (Load Case F3 in Table 3.1.3)

Under certain conditions, the fuel basket plates may be under direct compressive load. Although the finite element simulations can predict the onset of an instability and post-instability behavior, the computation in this subsection uses (the more conservative) classical instability formulations to demonstrate that an elastic instability of the basket plates is not credible.

A solution for the stability of the fuel basket plate is obtained using the classical formula for buckling of a wide bar [3.III.1]. Material properties are selected corresponding to a metal temperature of 325°C, which bounds the computed metal temperatures anywhere in the fuel basket (see Table 4.III.3). The critical buckling stress for a pin-ended bar is:

$$\sigma_{cr} = (\pi)^2 \frac{E}{12(1-\nu^2)}\left(\frac{h}{a}\right)^2$$

where h is the plate thickness, a is the unsupported plate length, E is the Young's Modulus of Metamic-HT at 325°C, ν is Poisson's Ratio (use 0.3 for this calculation)

TABLE 3.III.4
MAXIMUM DISPLACEMENT IN MPC-68M FUEL BASKET

Maximum Lateral Displacement in Fuel Basket Panel, θ (dimensionless) (Note 1)	Maximum Allowable Value of θ (from Table 2.III.4)	Safety Factor
8.9~~9.6~~ $\times 10^{-4}$	0.005	5.62~~5.21~~

Notes:

1. See Subsection 2.III.0.1 for definition of θ.

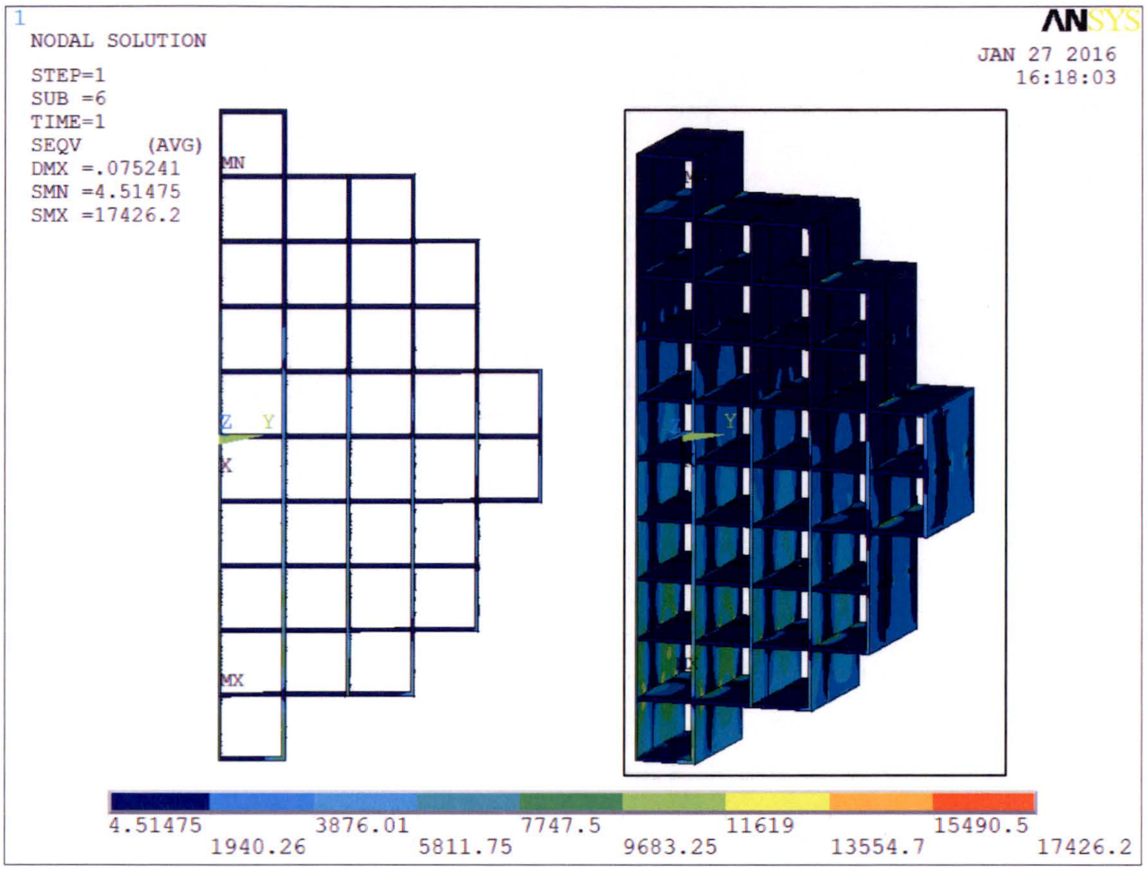

FIGURE 3.III.2: VON MISES STRESS DISTRIBUTION IN MPC-68M FUEL BASKET UNDER 60g LOAD

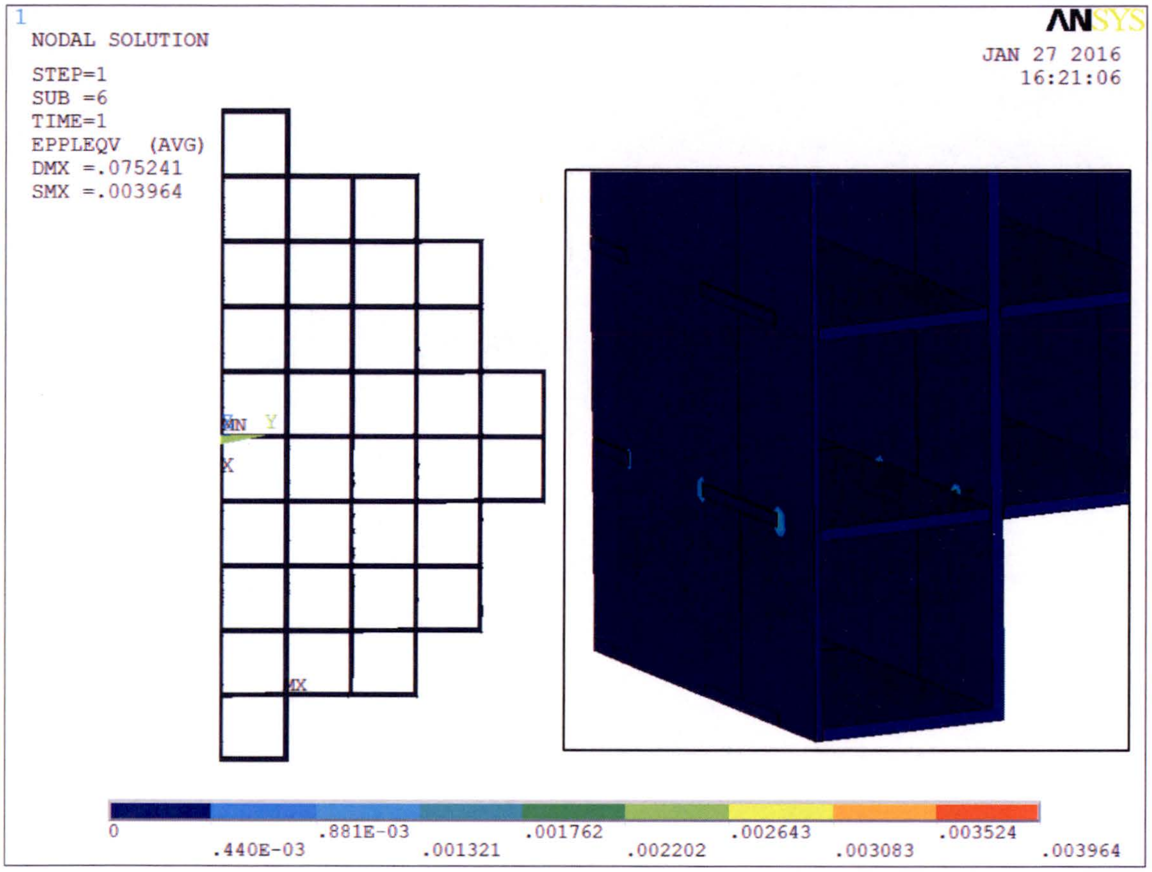

FIGURE 3.III.3: PLASTIC STRAIN DISTRIBUTION IN MPC-68M FUEL BASKET UNDER 60g LOAD

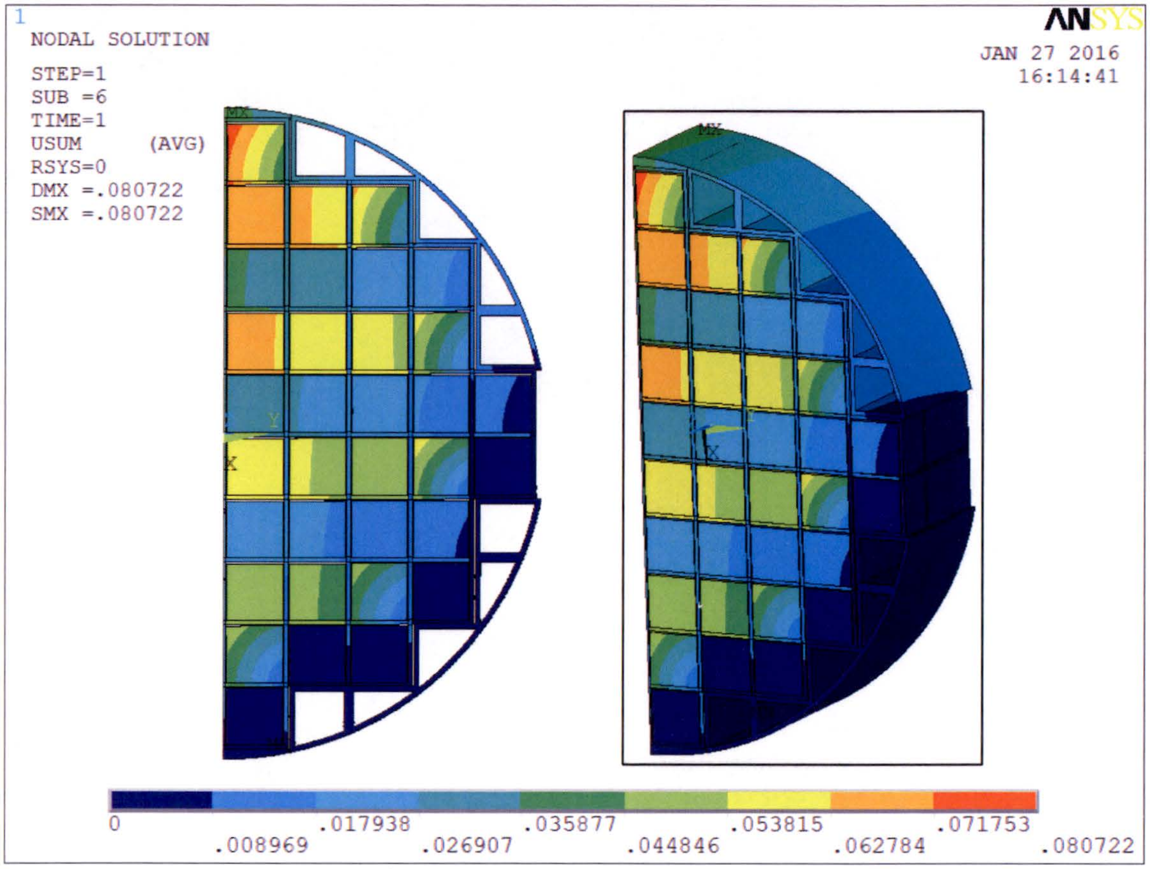

FIGURE 3.III.4: DISPLACEMENT CONTOURS IN MPC-68M FUEL BASKET UNDER 60g LOAD

CHAPTER 4[1] THERMAL EVALUATION

4.0 OVERVIEW

The HI-STORM System is designed for long-term storage of spent nuclear fuel (SNF) in a vertical orientation. An array of HI-STORM Systems laid out in a rectilinear pattern will be stored on a concrete ISFSI pad in an open environment. In this section, compliance of the HI-STORM thermal performance to 10CFR72 requirements for outdoor storage at an ISFSI is established. The analysis considers passive rejection of decay heat from the stored SNF assemblies to the environment under normal, off-normal, and accident conditions of storage. Effects of incident solar radiation (insolation) and partial radiation blockage due to the presence of neighboring casks at an ISFSI site are included in the analyses. 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 100 system are quantified. Safe thermal performance during on-site loading, unloading and transfer operations utilizing the HI-TRAC transfer cask is also demonstrated.

The HI-STORM thermal evaluation follows the guidelines of NUREG-1536 [4.4.1] and ISG-11 [4.1.4] to demonstrate thermal compliance of the HI-STORM system. . These guidelines provide specific limits on the permissible maximum cladding temperature in the stored commercial spent fuel (CSF)[2] 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 for long-term storage shall be limited to 752°F (400°C).

2. The fuel cladding temperature for short-term operations shall be limited to 752°F (400°C) for high burnup fuel and 1058°F (570°C) for moderate burnup fuel.

3. The fuel cladding temperature should be maintained below 1058°F (570°C) for accident and off-normal event conditions.

4. The maximum internal pressure of the MPC should remain within its design pressures for normal, off-normal, and accident conditions.

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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 (Table 1.0.1)~~.~~. ~~This chapter has been substantially re-written in support of LAR #3 to improve clarity and to incorporate the 3-D thermal model. Because of extensive editing a clean chapter is issued with this amendment.~~

2 Defined as nuclear fuel that is used to produce energy in a commercial nuclear reactor (See Table 1.0.1).

temperature. When the HI-TRAC transfer cask and the loaded MPC under water-flooded conditions is removed from the pool and staged in an ambient air environment, the water, MPC, and HI-TRAC transfer cask metal absorb the decay heat emitted by the fuel assemblies. This results in a slow temperature rise of the HI-TRAC transfer cask with time, starting from an initial pool water temperature. The rate of temperature rise is limited by the thermal inertia of the HI-TRAC transfer cask.

The available time before the water in the MPC would reach boiling is computed under a conservative set of assumptions summarized below:

i. Heat loss by natural convection and radiation from the exposed HI-TRAC surfaces to ambient air is neglected (i.e., an adiabatic heat-up calculation is performed).

ii. The smaller of the two (i.e., 100-ton and 125-ton) HI-TRAC transfer cask metal mass is credited in the analysis. The 100-ton design has a significantly smaller quantity of metal mass, which will result in a higher rate of temperature rise.

iii. The water mass in the MPC cavity is understated.

Table 4.5.2 summarizes the lower bound weights and thermal inertias of the constituent components in the loaded HI-TRAC transfer cask. The rate of temperature rise of the HI-TRAC transfer cask and contents during an adiabatic heat-up is given by the ratio Q/C where:

Q = Coincident fuel decay heat in the canister
C = Thermal inertia of a loaded HI-TRAC (Btu/°F) (See Table 4.5.2)

Therefore, the time-to-boil, τ is given by the simple algebraic formula $\tau = C(212-T)/Q$ where 212°F has been set as the boiling temperature and T represents the temperature of the pool water under fuel loading operations. The time-to-boil clock starts when the HI-TRAC is no longer submerged in the pool water. Table 4.5.3 provides a summary of τ at several representative heat loads and initial pool water temperatures. The calculation of time-to-boil for a loaded canister shall be made using the above formula. An alternate method using the FLUENT thermal model described in Section 4.5.1 can be adopted to evaluate the time for water within the MPC to boil using site-specific conditions. Principal modeling steps and acceptance criteria are defined in Table 4.5.10.

As set forth in the HI-STORM operating procedures, in the unlikely event that the maximum allowable time provided in Table 4.5.3 is found to be insufficient to complete wet transfer operations, a forced water circulation shall be initiated and maintained to remove the decay heat from the MPC cavity. In this case, relatively cooler water will enter via the MPC lid drain port connection and heated water will exit from the vent port. The minimum water flow rate required to maintain the MPC cavity water temperature below boiling with an adequate subcooling margin is determined as follows:

Table 4.5.10

PRINCIPAL SITE-SPECIFIC TIME-TO-BOIL MODELING STEPS

Step 1: Site Specific Conditions	Heat Loads Site Specific heat load map Ambient Temperature – Fuel handling building air temperature Initial Water Temperature – Candidate temperature defined by cask user HI-TRAC Insolation – None
Step 2: FLUENT Thermal Model	Incorporate HI-TRAC thermal methodologies (ii) thru (ix) defined in Section 4.5.1
Step 3: Run FLUENT Model	Apply thermal loads defined in Step 1 and compute the time dependent temperature field starting from the initial temperature defined in Step 1.
Step 4: Post-Process Results	Post-process FLUENT solution and obtain bulk water temperature $T_b(\tau)$ as a function of time τ. Interpolate $T_b(\tau)$ to compute maximum permissible time-to-boil τ^* meeting $T_b(\tau^*) < 212^oF$.

An alternate method using the FLUENT thermal model described in Section 4.4 can be adopted to evaluate HI-STORM site-specific fire accident event similar to that described in Section 4.6 of HI-STORM FW FSAR. Principal modeling steps and acceptance criteria are defined in Table 4.6.10.

(b) HI-TRAC Fire

During the handling of the HI-TRAC transfer cask, the transporter fuel tank capacity must be limited to a 50 gallons. The duration of the 50-gallon fire under the conservatively postulated spill defined in the HI-STORM fire evaluation computes as 4.775 minutes. To demonstrate the fuel cladding and MPC pressure boundary integrity under exposure to this fire duration event during a fire accident analysis of the loaded 100-ton HI-TRAC is performed. In this analysis, the contents of the HI-TRAC are conservatively postulated to undergo a transient heat-up as a lumped mass from the decay heat input and heat input from the short duration fire. This analysis, because of the lower mass of the 100-ton HI-TRAC, bounds the effects for the 125-ton HI-TRAC. Using understated thermal inertia of the HI-TRAC and design maximum heat load (36.9 kW) the temperature rise rate computes as 5.553°F/min. Therefore, the temperature rise computed as the product of this rate and the fire duration reported above is 26.5°F. In this manner the maximum cladding temperature obtained by adding the temperature rise to the initial condition (See Table 4.5.6 for design basis heat load with X=3) computes as 811°F. The maximum fire temperature computed in the conservative manner above remains below the 1058°F accident temperature limit (Table 4.3.1) by substantial margins.

The elevated temperatures as a result of the fire accident will cause the pressure in the water jacket to increase and the overpressure relief valves to vent steam to the atmosphere. Based on the fire heat input to the water jacket, 11% of the water in the water jacket is boiled off. However, it is conservatively assumed, for dose calculations, that all the water in the water jacket is lost. In the 125-ton HI-TRAC, which uses Holtite in the lids for neutron shielding, the elevated fire temperatures would cause the Holtite to exceed its design accident temperature limits. This condition is conservatively addressed by ignoring neutron shield in the accident dose calculations.

Due to the increased temperatures of the MPC during fire accident the internal MPC pressure increases. The fire accident pressure is computed assuming the MPC cavity temperature rises by the fire accident temperature rise computed in this section. The result is tabulated in Table 4.6.2. The fire accident MPC pressure is substantially below the accident pressure limit (Table 2.2.1).

An alternate method using the FLUENT thermal model described in Section 4.5 can be adopted to evaluate HI-TRAC site-specific fire accident event. Principal modeling steps and acceptance criteria are defined in Table 4.6.11.

4.6.2.2 Jacket Water Loss

In this subsection, the fuel cladding and MPC boundary integrity is evaluated for a postulated loss of water from the HI-TRAC water jacket. The HI-TRAC is equipped with an array of water compartments filled with water. For a bounding analysis, all water compartments are assumed to

be impervious to air. Using this model, a transient thermal solution of the HI-STORM 100 System starting from normal storage conditions is obtained. The results of the blocked ducts transient analysis are presented in Table 4.6.5 and confirmed to be below the accident temperature limits (Table 2.2.3). The co-incident MPC pressure is also computed and compared with the accident design pressure (Table 2.2.1). The result (Table 4.6.2) is confirmed to be below the limit.

For MPC heat loads which meet the values in Table 4.5.7 or 4.5.8, the results of the transient analysis that support the required action completion times for clearing the inlets are presented in Table 4.6.7 and confirm all temperatures are below the accident temperature limits (Table 2.2.3).

As noted above, the fuel and component temperatures rise due to complete blockage of HI-STORM vents. This temperature rise is small for casks where heat loads are much lower than design basis heat loads. A threshold heat load is defined in Table 4.6.8 at or below which fuel and component temperatures remain below their accident temperature limits under steady state conditions. A steady state evaluation of a complete vent blockage at this threshold heat load is performed for MPC-68 and results presented in Table 4.6.9. The results demonstrate that the fuel and component temperatures remain below their accident temperature limits defined in the Design Criteria Chapter 2 with robust margins. To identify and clear any blockages mandatory surveillance is defined in Chapter 11. Since the thermal performance of MPC-68 is essentially the same as MPC-32 or bounds the other MPC types as demonstrated in Section 4.4, the threshold heat load in Table 4.6.8 is adopted for all MPC types.

4.6.2.5 Burial Under Debris

Burial of the HI-STORM 100 System under debris is not a credible accident. During storage at the ISFSI there are no structures over the casks. 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 100 System to become completely buried under debris. However, for conservatism, complete burial under debris is considered.

To demonstrate the inherent safety of the HI-STORM 100 System, a bounding analysis that considers the debris to act as a perfect insulator is considered. Under this scenario, the contents of the HI-STORM 100 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 100 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:

Table 4.6.8

THRESHOLD DECAY HEAT FOR 100% VENT BLOCKAGE

MPC Threshold Decay Heat	19 kW
Note: The heat load at any storage location in the basket must be less than or equal to the threshold heat load tabulated herein divided by the number of storage locations.	

Table 4.6.9

STEADY STATE MAXIMUM HI-STORM TEMPERATURES AT THRESHOLD HEAT LOAD UNDER 100% VENT BLOCKAGE

Component	**Temperatures (^{o}F)**
Fuel Cladding	716
MPC Basket	712
MPC Shell	482
Overpack Inner Shell	407
Overpack Concrete Section Temperature	270
Lid Concrete Bottom Plate	375
Lid Concrete Section Temperature	293

Table 4.6.10

PRINCIPAL SITE-SPECIFIC HI-STORM FIRE ACCIDENT MODELING STEPS

Step 1: Site Specific Conditions	Heat Loads Site Specific heat load map. Ambient Temperature – Normal storage temperature defined in Chapter 2. Fire Accident – Compute fire duration τ_f based on site specific fuel quantity in accordance with methodology defined in Sub-Section 4.6.2.1(a).
Step 2: FLUENT Thermal Model	Incorporate HI-STORM thermal methodologies defined in Sub-Sections 4.4.1.1 and 4.4.1.2. Apply heat loads and ambient temperature defined in Step 1 and obtain baseline initial temperature field.
Step 3: Fire Transient Solution	Apply fire parameters defined by fire temperature, fire emissivity and convection heat transfer coefficient specified in Sub-Section 4.6.2.1(a) to FLUENT Model and compute time dependent HI-STORM temperature field starting from initial temperature field obtained in Step 2 upto end of fire τ_f.
Step 4: Post-Fire Solution	Restore ambient temperature conditions as defined in Sub-Section 4.6.2.1(a) and compute time dependent temperature field under cooldown of HI-STORM cask by natural convection and radiation. Conservatively assume paint loss from all exterior surfaces. Continue solution until all component and fuel temperatures reach their maximum and begin to recede.
Step 5: Post-Process Results	Post-process FLUENT solution and evaluate compliance of maximum fuel, basket, MPC confinement boundary, HI-STORM concrete and enclosure shell temperatures to Chapter 2 accident temperature limits. Compute maximum MPC pressure in accordance with Sub-Section 4.4.5 methodology and evaluate compliance with Chapter 2 accident pressure limits.

Table 4.6.11

PRINCIPAL SITE-SPECIFIC HI-TRAC FIRE ACCIDENT MODELING STEPS

Step 1: Site Specific Conditions	Heat Loads Site Specific heat load map. Ambient Temperature – Short Term Operations temperature defined in Chapter 2. Fire Accident – Compute fire duration τ_f based on site specific fuel quantity in accordance with methodology defined in Sub-Section 4.6.2.1(b).
Step 2: FLUENT Thermal Model	Incorporate HI-TRAC thermal methodologies (i) thru (ix) defined in Section 4.5.1. Apply heat loads and ambient temperature defined in Step 1 and obtain baseline initial temperature field.
Step 3: Fire Transient Solution	Apply fire parameters defined by fire temperature, fire emissivity and convection heat transfer coefficient specified in Sub-Section 4.6.2.1(a) to FLUENT Model and compute time dependent HI-TRAC temperature field starting from initial temperature field obtained in Step 2 upto end of fire τ_f.
Step 4: Post-Fire Solution	Restore ambient temperature conditions as defined in Sub-Section 4.6.2.1(a). Conservatively assume destruction of paint from exterior surfaces and complete Holtite loss. Compute time dependent temperature field under cooldown of HI-TRAC cask by natural convection and radiation. Continue solution until all component and fuel temperatures reach their maximum and begin to recede.
Step 5: Post-Process Results	Post-process FLUENT solution and evaluate compliance of maximum fuel, basket, MPC confinement boundary and HI-TRAC enclosure shell temperatures with Chapter 2 accident temperature limits. Compute maximum MPC pressure in accordance with Sub-Section 4.4.5 methodology and evaluate compliance with Chapter 2 accident pressure limits.

- Cladding temperature is substantially below the ISG-11, Rev. 3 limit.
- MPC basket is below the design limit (Table 4.III.2) by large margin.
- MPC shell and Overpack metal temperatures are below design limits (Table 2.2.3).
- Overpack body and lid concrete are well below design limits (Table 4.3.1).

4.III.5 THERMAL EVALUATION OF SHORT TERM OPERATIONS

4.III.5.1 HI-TRAC Thermal Model

The HI-TRAC thermal model presented in Section 4.5 is adopted for the evaluation of MPC-68M under short term operations.

4.III.5.2 Maximum Time Limit During Wet Transfer Operations

As the MPC thermal inertia credited in the time-to-boil calculations is bounded by the MPC-68M thermal inertia the evaluation of wet transfer operations in Section 4.5 remains applicable to the MPC-68M. An alternate method using the FLUENT thermal model described in Section 4.III.5.1 can be adopted to evaluate the time for water within the MPC to boil using site-specific conditions. Principal modeling steps and acceptance criteria are defined in Table 4.5.10.

4.III.5.3 MPC Temperature During Moisture Removal Operations

4.III.5.3.1 Vacuum Drying

Prior to helium backfill the MPC-68M must be drained of water and demoisturized. At the start of draining operation, both the HI-TRAC 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 region is uncovered during the draining operation, the fuel and basket mass will undergo a monotonic heat up from the initially cold conditions when the heated surfaces were submerged under water. To limit fuel temperatures demoisturization of the MPC-68M by the vacuum drying method is permitted provided the HI-TRAC annulus remains water filled during vacuum drying operations. To support vacuum drying operations two limiting scenarios are defined below:

Scenario A: The MPC-68M is loaded with Moderate Burnup Fuel assemblies generating heat at the maximum permissible rate defined in Chapter 2 under the bounding regionalized storage scenario X = 0.5.

Scenario B: The MPC-68M is loaded with one or more High Burnup Fuel assemblies and the MPC-68M decay heat is less than a conservatively defined threshold heat load Q = 29 kW[1].

1 Threshold heat load is defined as the product of maximum loaded assembly heat load r_{max} and the number of fuel storage cells (n=68). Under this stipulation r_{max} must not exceed 0.426 kW.

The time constant of the cask body (i.e., the overpack) can be determined using the formula:

$$\tau = \frac{c_p \times \rho \times L_c^2}{k}$$

where:

c_p = Overpack Specific Heat Capacity (Btu/lb-°F)
ρ = Overpack Density (lb/ft^3)
L_c = Overpack Characteristic Length (ft)
k = Overpack Thermal Conductivity (Btu/ft-hr-°F)

The concrete contributes the majority of the overpack mass and volume, so we will use the specific heat capacity (0.156 Btu/lb-°F), density (140 lb/ft^3) and thermal conductivity (1.05 Btu/ft-hr-°F) of concrete for the time constant calculation. The characteristic length of a hollow cylinder is its wall thickness. The characteristic length for the HI-STORM Overpack is therefore 29.5 in, or approximately 2.46 ft. Substituting into the equation, the overpack time constant is determined as:

$$\tau = \frac{0.156 \times 140 \times 2.46^2}{1.05} = 126\, hrs$$

One-tenth of this time constant is approximately 12.6 hours (756 minutes), substantially longer than the fire duration of 3.62 minutes, so the MPC is evaluated by considering the MPC canister as an adiabatic boundary. The fuel temperature rise is computed next.

Table 4.III.10 lists lower-bound thermal inertia values for the MPC-68M and the contained fuel assemblies. Applying design heat load (36.9 kW (1.26x10^5 Btu/hr)) and adiabatic heating for the 3.62 minutes fire, the fuel temperature rise computes as:

$$\Delta T_{fuel} = \frac{\text{Decay heat} \times \text{Time duration}}{(\text{MPC} + \text{Basket \& Shims} + \text{Fuel}) \text{ heat capacities}} = \frac{1.26 \times 10^5 \text{ Btu/hr} \times (3.62/60) hr}{(2400 + 2339 + 2780) \text{ Btu/}^\circ F} = 1.0^\circ F$$

This is a very small increase in fuel temperature. Consequently, the impact on the MPC internal helium pressure will be quite small. Based on a conservative analysis of the HI-STORM 100 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. We conclude that the ability of the HI-STORM 100 System to cool the spent nuclear fuel within design temperature limits during and after fire is not compromised.

An alternate method using the FLUENT thermal model described in Section 4.III.4 can be adopted to evaluate HI-STORM site-specific fire accident event similar to that described in Section 4.6 of HI-STORM FSAR. Principal modeling steps and acceptance criteria are defined in Table 4.6.10.

(ii) HI-TRAC Fire[1]

To demonstrate the fuel cladding and MPC pressure boundary integrity under an exposure to a hypothetical short duration fire event during on-site handling operations, a fire accident analysis of the loaded 100-ton HI-TRAC is performed. This analysis, because of the lower mass of the 100-ton HI-TRAC, bounds the effects for the 125-ton HI-TRAC. In this analysis, the contents of the HI-TRAC are conservatively postulated to undergo a transient heat-up as a lumped mass from the decay heat input and heat input from the short duration fire. The rate of temperature rise of the HI-TRAC depends on the thermal inertia of the cask, the cask initial conditions, the spent nuclear fuel decay heat generation, and the fire heat flux. Using conservatively bounding inputs – lowerbound thermal inertia, steady state maximum cask temperatures (Table 4.III.6) and design heat load (36.9 kW) - a bounding cask temperature rise of 5.178°F per minute is computed from the combined radiant and forced convection fire and decay heat inputs to the cask. During the handling of the HI-TRAC transfer cask, the transporter is limited to a maximum of 50 gallons. The duration of the 50-gallon fire using the methodology articulated above for HI-STORM fire is 4.775 minutes. Therefore, the temperature rise computed as the product of the rate of temperature rise and the fire duration is 24.7°F, and the co-incident fuel cladding temperature (664.7°F)[2] is below the 1058°F accident limit.

The elevated temperatures as a result of the fire accident will cause the pressure in the water jacket to increase and cause the overpressure relief valves to vent steam to the atmosphere. Based on the fire heat input to the water jacket, less than 11% of the water in the water jacket can be boiled off. However, it is conservatively assumed, for dose calculations, that all the water in the water jacket is lost. In the 125-ton HI-TRAC, which uses Holtite in the lids for neutron shielding, the elevated fire temperatures would cause the Holtite to exceed its design accident temperature limits. It is conservatively assumed, for dose calculations, that all the Holtite in the 125-ton HI-TRAC is lost.

Due to the increased temperatures the MPC experiences as a result of the fire accident in the HI-TRAC transfer cask, the MPC internal pressure increases. The pressure rise is computed using the Ideal Gas Law and upperbound helium backfill pressure defined in Chapter 4, Table 4.4.12 and results tabulated in Table 4.III.9. The computed MPC accident pressure is substantially below the accident design pressure (Table 2.2.1).

An alternate method using the FLUENT thermal model described in Section 4.III.5 can be adopted to evaluate HI-TRAC site-specific fire accident event. Principal modeling steps and acceptance criteria are defined in Table 4.6.11.

(b) Flood

The flood accident is defined in Chapter 2 as a deep submergence event. The worst flood from a thermal perspective is a "smart flood" that just rises to the top of the inlets to prevent airflow without

1 The HI-TRAC fire accident methodology is same as the generic methodology in Section 4.6 of the HI-STORM 100 FSAR.

2 Computed by adding the fire temperature rise to initial fuel temperature (Table 4.III.6).

the benefit of MPC cooling by water. This effect is bounded by the 100% inlets ducts blockage accident evaluated herein in Section 4.III.6.2(d).

(c) Burial Under Debris

The burial under debris evaluation in Section 4.6 is bounding because of the following:

(i) The MPC thermal inertia is neglected.
(ii) The initial storage temperatures under MPC-68M storage are less than the HI-STORM 100 System temperatures.

(d) 100% Blockage of Air Ducts

This accident is defined in Section 4.6 as 100% blockage of the air inlet ducts for 32 hours. This event is evaluated by blocking the air inlets in the FLUENT thermal model and computing the 32-hour temperature rise of the MPC and stored fuel. The results of this analysis are tabulated in Table 4.III.7. The results show that fuel cladding and component temperatures remain below their respective accident limits specified in Chapter 2 and Supplement 4.III. The increase in temperature results in a concomitant rise of the MPC pressure. The maximum accident pressure tabulated in Table 4.III.7 is below the design limit specified in Chapter 2.

Since the temperatures of MPC-68M are bounded by the MPCs evaluated in Chapter 4, threshold heat load defined in Table 4.6.8 can also be adopted for MPC-68M. A threshold heat load is defined in Table 4.6.8 at or below which periodic surveillance or vent blockage corrective actions defined in Sub-Section 11.2.13.2 are applicable.

(e) Extreme Environmental Temperature

The principal effect of elevated ambient temperature is a rise of the HI-STORM 100 temperatures from the baseline normal storage temperatures by the difference between elevated ambient and normal ambient temperatures. As the normal storage temperatures under MPC-68M storage in the HI-STORM 100 overpack are bounded by the HI-STORM 100 System temperatures reported in Section 4.4, the temperatures under this event are likewise bounded by the extreme ambient evaluation in Section 4.6.

(f) 100% Rods Rupture Accident

In accordance with NUREG-1536 a 100% rods rupture accident is evaluated assuming 100% of the rods fill gases and fission gases release in accordance with NUREG-1536 release fractions. The MPC-68M pressure under this postulated accident is computed and tabulated in Table 4III.4. The pressure is below the accident design pressure (Table 2.2.1).

(g) Jacket Water Loss

CHAPTER 5†: SHIELDING EVALUATION

5.0 INTRODUCTION

The shielding analysis of the HI-STORM 100 System, including the HI-STORM 100 overpack, HI-STORM 100S overpack, HI-STORM 100S Version B overpack††, and the 100-ton (including the 100D) and 125-ton (including the 125D) HI-TRAC transfer casks, is presented in this chapter. The HI-STORM 100 System is designed to accommodate different MPCs within HI-STORM overpacks (the HI-STORM 100S overpack is a shorter version of the HI-STORM 100 overpack and the HI-STORM 100S Version B is shorter than both the HI-STORM 100 and 100S overpacks). The MPCs are designated as MPC-24, MPC-24E and MPC-24EF (24 PWR fuel assemblies), MPC-32 and MPC-32F (32 PWR fuel assemblies), and MPC-68, MPC-68F, and MPC-68FF (68 BWR fuel assemblies). The MPC-24E and MPC-24EF are essentially identical to the MPC-24 from a shielding perspective. Therefore only the MPC-24 is analyzed in this chapter. Likewise, the MPC-68, MPC-68F and MPC-68FF are identical from a shielding perspective as are the MPC-32 and MPC-32F and therefore only the MPC-68 and MPC-32 are analyzed. Throughout this chapter, unless stated otherwise, MPC-24 refers to either the MPC-24, MPC-24E, or MPC-24EF and MPC-32 refers to either the MPC-32 or MPC-32F and MPC-68 refers to the MPC-68, MPC-68F, and MPC-68FF.

In addition to storing intact PWR and BWR fuel assemblies, the HI-STORM 100 System is designed to store BWR and PWR damaged fuel assemblies and fuel debris. Damaged fuel assemblies and fuel debris are defined in Sections 2.1.3 and 2.1.9. Both damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs).

The MPC-68, MPC-68F, ~~and~~ MPC-68FF, and MPC-68M are also capable of storing Dresden Unit 1 antimony-beryllium neutron sources and the single Thoria rod canister which contains 18 thoria rods that were irradiated in two separate fuel assemblies.

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† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in *Chapter 1,* Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

†† The HI-STORM 100S Version B was implemented in the HI-STORM FSAR (between Revisions 2 and 3) through the 10 CFR 72.48 process. The discussion of the HI-STORM 100S Version B and associated results were added to LAR 1014-2 at the end of the review cycle to support the NRC review of the radiation protection program proposed in the Certificate of Compliance in LAR 1014-2. The NRC did not review and approve any aspect of the design of the HI-STORM 100S Version B since it has been implemented under the provisions of 10 CFR 72.48.

5.1 DISCUSSION AND RESULTS

The principal sources of radiation in the HI-STORM 100 System are:

- Gamma radiation originating from the following sources

    1. Decay of radioactive fission products
    2. Secondary photons from neutron capture in fissile and non-fissile nuclides
    3. Hardware activation products generated during core operations

- Neutron radiation originating from the following sources

    1. Spontaneous fission
    2. α,n reactions in fuel materials
    3. Secondary neutrons produced by fission from subcritical multiplication
    4. γ,n reactions (this source is negligible)
    5. Dresden Unit 1 antimony-beryllium neutron sources

During loading, unloading, and transfer operations, shielding from gamma radiation is provided by the steel structure of the MPC and the steel, lead, and water of the HI-TRAC transfer cask. For storage, the gamma shielding is provided by the MPC, and the steel and concrete 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. Additionally, in the HI-TRAC 125 and 125D top lid and the transfer lid of the HI-TRAC 125, a solid neutron shielding material, Holtite-A is used to thermalize the neutrons. Boron carbide, dispersed in the solid neutron shield material utilizes the high neutron absorption cross section of ^{10}B to absorb the thermalized neutrons.

The shielding analyses were performed with MCNP-4A [5.1.1] 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 4.3 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 B&W 15x15 and the GE 7x7, for PWR and BWR fuel types, respectively. The design basis intact 6x6 and mixed oxide (MOX) fuel assemblies are the GE 6x6. The GE 6x6 is also the design basis damaged fuel assembly for the Dresden Unit 1 and Humboldt Bay array classes. Section 2.1.9 specifies the acceptable intact zircaloy clad fuel characteristics and the acceptable damaged fuel characteristics.

The design basis stainless steel clad fuels are the WE 15x15 and the A/C 10x10, for PWR and BWR fuel types, respectively. Section 2.1.9 specifies the acceptable fuel characteristics of stainless steel clad fuel for storage.

The MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-32F, MPC-68, and MPC-68FF are qualified for storage of SNF with different combinations of maximum burnup levels and minimum cooling times. Section 2.1.9 specifies the acceptable maximum burnup levels and minimum cooling times for storage of zircaloy clad fuel in these MPCs. Section 2.1.9 also specifies the acceptable maximum burnup levels and minimum cooling times for storage of stainless steel clad fuel. ~~The burnup and cooling time values in Section 2.1.9, which differ by array class, were chosen based on an analysis of the maximum decay heat load that could be accommodated within each MPC. Section 5.2 of this chapter describes the choice of the design basis fuel assembly based on a comparison of source terms and also provides a description of how the allowable burnup and cooling times were derived. Since for a given cooling time, different array classes have different allowable burnups in Section 2.1.9, burnup and cooling times that bound array classes 14x14A and 9x9G were used for the analysis in this chapter since these array class burnup and cooling time combinations bound the combinations from the other PWR and BWR array classes. Section 5.2.5 describes how this results in a conservative estimate of the maximum dose rates.~~

Section 2.1.9 specifies that the maximum assembly average burnup for PWR and BWR fuel is 68,200 and 65,000 MWD/MTU, respectively. The analysis in this chapter conservatively considers burnups up to 75,000 and 70,000 MWD/MTU for PWR and BWR fuel, respectively.

The burnup and cooling time combinations listed below bound all acceptable uniform and regionalized loading burnup levels and cooling times from Section 2.1.9. All combinations were analyzed in the HI-STORM overpack and HI-TRAC transfer casks.

Zircaloy Clad Fuel		
MPC-24	**MPC-32**	**MPC-68**
60,000 MWD/MTU 3 year cooling	45,000 MWD/MTU 3 year cooling	50,000 MWD/MTU 3 year cooling
69,000 MWD/MTU 4 year cooling	60,000 MWD/MTU 4 year cooling	62,000 MWD/MTU 4 year cooling
75,000 MWD/MTU 5 year cooling	69,000 MWD/MTU 5 year cooling	65,000 MWD/MTU 5 year cooling
		70,000 MWD/MTU 6 year cooling

The initial enrichment used in the analysis is consistent with Table 5.2.24. The results of the comparison are provided in Table 5.2.27. These results indicate that the B&W 15x15 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 2.1.1. This fuel assembly also has the highest UO_2 mass (see Table 5.2.25) which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO_2 mass produces the highest radiation source term. The power/assembly values used in Table 5.2.25 were calculated by dividing 110% of the thermal power for commercial PWR reactors using that array class by the number of assemblies in the core. The higher thermal power, 110%, was used to account for potential power uprates. The power level used for the B&W15 is an additional 17% higher for consistency with previous revisions of the FSAR which also used this assembly as the design basis assembly.

The Haddam Neck and San Onofre 1 classes are shorter stainless steel clad versions of the WE 15x15 and WE 14x14 classes, respectively. Since these assemblies have stainless steel clad, they were analyzed separately as discussed in Section 5.2.3. Based on the results in Table 5.2.27, which show that the WE 15x15 assembly class has a higher source term than the WE 14x14 assembly class, the Haddam Neck, WE 15x15, fuel assembly was analyzed as the bounding PWR stainless steel clad fuel assembly. The Indian Point 1 fuel assembly is a unique 14x14 design with a smaller mass of fuel and clad than the WE14x14. Therefore, it is also bounded by the WE 15x15 stainless steel fuel assembly.

~~As discussed below in Section 5.2.5.3, the allowable burnup limits in Section 2.1.9 were calculated for different array classes rather than using the design basis assembly to calculate the allowable burnups for all array classes. As mentioned above, the design basis assembly has the highest neutron and gamma source term of the various array classes for the same burnup and cooling time. In order to account for the fact that different array classes have different allowable burnups for the same cooling time, burnups which bound the 14x14A array class were used with the design basis assembly for the analysis in this chapter because those burnups bound the burnups from all other PWR array classes. This approach assures that the calculated source terms and dose rates will be conservative.~~

5.2.5.2 <u>BWR Design Basis Assembly</u>

Table 2.1.2 lists the BWR fuel assembly classes that were evaluated to determine the design basis BWR fuel assembly. Since there are minor differences between the array types in the GE BWR/2-3 and GE BWR/4-6 assembly classes, these assembly classes were not considered individually but rather as a single class. Within that class, the array types, 7x7, 8x8, 9x9, and 10x10 were analyzed to determine the bounding BWR fuel assembly. Since the Humboldt Bay 7x7 and Dresden 1 8x8 are smaller versions of the 7x7 and 8x8 assemblies they are bounded by the 7x7 and 8x8 assemblies in the GE BWR/2-3 and GE BWR/4-6 classes. Within each array type, the fuel assembly with the highest UO_2 mass was analyzed. Since the variations of fuel assemblies within an array type are very minor, it is conservative to choose the assembly with the highest UO_2 mass. For a given array type of assemblies, the one with the highest UO_2 mass will produce the highest radiation source because, for a given burnup (MWD/MTU) and enrichment,

it will have produced the most energy and therefore the most fission products. The Humboldt Bay 6x6, Dresden 1 6x6, and LaCrosse assembly classes were not considered in the determination of the bounding fuel assembly. However, these assemblies were analyzed explicitly as discussed below.

Table 5.2.26 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad BWR fuel assembly. The corresponding fuel assembly array class from Section 2.1.9 is also listed in the table. The fuel assembly listed for each array type is the assembly that has the highest UO_2 mass. All fuel assemblies in Table 5.2.26 were analyzed at the same burnup and cooling time. The initial enrichment used in these analyses is consistent with Table 5.2.24. The results of the comparison are provided in Table 5.2.28. These results indicate that the 7x7 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 2.1.2. This fuel assembly also has the highest UO_2 mass which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO_2 mass produces the highest radiation source term. According to Reference [5.2.6], the last discharge of a 7x7 assembly was in 1985 and the maximum average burnup for a 7x7 during their operation was 29,000 MWD/MTU. This clearly indicates that the existing 7x7 assemblies have an average burnup and minimum cooling time that is well within the burnup and cooling time limits in Section 2.1.9. Therefore, the 7x7 assembly has never reached the burnup level analyzed in this chapter. However, in the interest of conservatism the 7x7 was chosen as the bounding fuel assembly array type. The power/assembly values used in Table 5.2.26 were calculated by dividing 120% of the thermal power for commercial BWR reactors by the number of assemblies in the core. The higher thermal power, 120%, was used to account for potential power uprates. The power level used for the 7x7 is an additional 4% higher for consistency with previous revisions of the FSAR which also used this assembly as the design basis assembly.

Since the LaCrosse fuel assembly type is a stainless steel clad 10x10 assembly it was analyzed separately. The maximum burnup and minimum cooling time for this assembly are limited to 22,500 MWD/MTU and 10-year cooling as specified in Section 2.1.9. This assembly type is discussed further in Section 5.2.3.

The Humboldt Bay 6x6 and Dresden 1 6x6 fuel are older and shorter fuel than the other array types analyzed and therefore are considered separately. The Dresden 1 6x6 was chosen as the design basis fuel assembly for the Humboldt Bay 6x6 and Dresden 1 6x6 fuel assembly classes because it has the higher UO_2 mass. Dresden 1 also contains a few 6x6 MOX fuel assemblies, which were explicitly analyzed as well.

Reference [5.2.6] indicates that the Dresden 1 6x6 fuel assembly has a higher UO_2 mass than the Dresden 1 8x8 or the Humboldt Bay fuel (6x6 and 7x7). Therefore, the Dresden 1 6x6 fuel assembly was also chosen as the bounding assembly for damaged fuel and fuel debris for the Humboldt Bay and Dresden 1 fuel assembly classes.

Since the design basis 6x6 fuel assembly can be intact or damaged, the analysis presented in Section 5.4.2 for the damaged 6x6 fuel assembly also demonstrates the acceptability of storing intact 6x6 fuel assemblies from the Dresden 1 and Humboldt Bay fuel assembly classes.

~~As discussed below in Section 5.2.5.3, the allowable burnup limits in Section 2.1.9 were calculated for different array classes rather than using the design basis assembly to calculate the allowable burnups for all array classes. As mentioned above, the design basis assembly has the highest neutron and gamma source term of the various array classes for the same burnup and cooling time. In order to account for the fact that different array classes have different allowable burnups for the same cooling time, burnups which bound the 9x9G array class were used with the design basis assembly for the analysis in this chapter because those burnups bound the burnups from all other BWR array classes. This approach assures that the calculated source terms and dose rates will be conservative.~~

5.2.5.3 Deleted ~~Decay Heat Loads and Allowable Burnup and Cooling Times~~

~~Section 2.1.6 describes the calculation of the MPC maximum decay heat limits per assembly. These limits, which differ for uniform and regionalized loading, are presented in Section 2.1.9. The allowable burnup and cooling time limits are derived based on the allowable decay heat limits. Since the decay heat of an assembly will vary slightly with enrichment for a fixed burnup and cooling time, an equation is used to represent burnup as a function of decay heat and enrichment. This equation is of the form:~~

$$B_u = A*q + B*q^2 + C*q^3 + D*E_{235}{}^2 + E*E_{235}*q + F*E_{235}*q^2 + G$$

~~where:~~
~~B_u = Burnup in MWD/MTU~~
~~q = assembly decay heat (kW)~~
~~E_{235} = wt.% ^{235}U~~

~~The coefficients for this equation were developed by fitting ORIGEN-S calculated data for a specific cooling time using GNUPLOT [5.2.16]. ORIGEN-S calculations were performed for enrichments ranging from 0.7 to 5.0 wt.% ^{235}U and burnups from 10,000 to 65,000 MWD/MTU for BWRs and 10,000 to 70,000 MWD/MTU for PWRs. The burnups were increased in 2,500 MWD/MTU increments. Using the ORIGEN-S data, the coefficients A through G were determined and then the constant, G, was adjusted so that all data points were bounded (i.e. calculated burnup less than or equal to ORIGEN-S value) by the fit. The coefficients were calculated using ORIGEN-S data for cooling times from 3 years to 20 years. As a result, Section 2.1.9 provides different equation coefficients for each cooling time from 3 to 20 years. Additional discussion on the determination of the equation coefficients is provided in Appendix 5.F. Since the decay heat increases as the enrichment decreases, the allowable burnup will decrease as the enrichment decreases. Therefore, the enrichment used to calculated the allowable burnups becomes a minimum enrichment value and assemblies with an enrichment higher than the value used in the equation are acceptable for storage assuming they also meet the~~

corresponding burnup and decay heat requirements. Even though the lower limit of 0.7 wt.% ^{235}U was used in developing the coefficients, these equations are valid for the few assemblies that might exist with enrichments below 0.7 wt.% ^{235}U. This is because the curve fit is very well behaved in the enrichment range from 0.7 to 5.0 wt.% ^{235}U and, therefore, it is expected that the curve fit will remain accurate for enrichments below 0.7 wt.% ^{235}U.

Different array classes or combinations of classes were analyzed separately to determine the allowable burnup as a function of cooling time for the specified allowable decay heat limits. Calculating allowable burnups for individual array classes is appropriate because even two assemblies with the same MTU may have a different allowable burnup for the same allowable cooling time and permissible decay heat. The heavy metal mass specified in Table 5.2.25 and 5.2.26 and Section 2.1.9 for the various array classes is the value that was used in the determination of the coefficients as a function of cooling time and is the maximum for the respective assembly class. Equation coefficients for each array class listed in Tables 5.2.25 and 5.2.26 were developed. In the end, the equation for the 17x17B and 17x17C array classes resulted in almost identical burnups. Therefore, in Section 2.1.9 these array classes were combined and the coefficients for the 17x17C array class were used since these coefficients produce slightly lower allowable burnups.

There is some uncertainty associated with the ORIGEN-S calculations due to uncertainty in the physics data (e.g. cross sections, decay constants, etc.) and the modeling techniques. To estimate this uncertainty, an approach similar to the one in Reference [5.2.14] was used. As a result, the potential error in the ORIGEN-S decay heat calculations was estimated to be in the range of 3.5 to 5.5% at 3 year cooling time and 1.5 to 3.5% at 20 year cooling. The difference is due to the change in isotopes important to decay heat as a function of cooling time. In order to be conservative in the derivation of the coefficients for the burnup equation, a 5% decay heat penalty was applied for both the PWR and BWR array classes.

As a demonstration that the decay heat values used to determine the allowable burnups are conservative, a comparison between these calculated decay heats and the decay heats reported in Reference [5.2.7] are presented in Table 5.2.29. This comparison is made for a burnup of 30,000 MWD/MTU and a cooling time of 5 years. The burnup was chosen based on the limited burnup data available in Reference [5.2.7].

As mentioned above, the fuel assembly burnup and cooling times in Section 2.1.9 were calculated using the decay heat limits which are also stipulated in Section 2.1.9. The burnup and cooling times for the non-fuel hardware, in Section 2.1.9, were chosen based on the radiation source term calculations discussed previously. The fuel assembly burnup, decay heat, and enrichment equations were derived without consideration for the decay heat from BPRAs, TPDs, CRAs, or APSRs. This is acceptable since the user of the HI-STORM 100 system is required to demonstrate compliance with the assembly decay heat limits in Section 2.1.9 regardless of the heat source (assembly or non-fuel hardware) and the actual decay heat from the non-fuel hardware is expected to be minimal. In addition, the shielding analysis presented in this chapter conservatively calculates the dose rates using both the burnup and cooling times for the fuel

~~assemblies and non-fuel hardware. Therefore, the safety of the HI-STORM 100 system is guaranteed through the bounding analysis in this chapter, represented by the burnup and cooling time limits in the CoC, and the bounding thermal analysis in Chapter 4, represented by the decay heat limits in the CoC.~~

5.2.5.4 Burnup, Enrichment and Cooling time values for Site Specific Dose Analyses

As discussed earlier in this Chapter, site-specific dose evaluations are required to show compliance with the regulatory requirements, and those need to consider the types, burnups, enrichments and cooling times of the fuel to be stored. Since it is impractical to evaluate every fuel assembly individually, a bounding approach is typically used where assemblies are grouped and bounding characteristics are selected and evaluated for each group. Recommendations and guidance for those selections are as follows:

For the fuel assembly type, the one approach would be to use the design basis assembly type, since this has been shown to bound all other assembly types (see Subsections 5.2.5.1 and 5.2.5.2). However, if this approach is considered too conservative, it is also acceptable to utilize a site-specific fuel assembly type. In this case, that fuel assembly type needs to be considered in both the radiation transport analyses and the source term evaluations.

For burnups, enrichments and cooling times, selecting an appropriate burnup and enrichment combination or combinations (for given lower bound cooling times) could be difficult, since the more conservative values are a higher burnup but a lower enrichment (see Subsection 5.2.2). One approach would be to have a single group, i.e. select bounding values for all those parameters: upper bound burnup, lower bound enrichment, and lower bound cooling time. However, this could be excessively conservative, since combinations of high burnup and low enrichment are typically not found in spent fuel. A more practical approach would be to establish several groups, each with an upper bound burnup and lower bound enrichment. A separate evaluation may be required for each group in that case. However, with the help of the information shown in Table 5.2.24 it may be possible to avoid multiple analyses and demonstrate compliance with a single set of burnup, enrichment and cooling time. To support this approach, the results of source term calculation for all burnup and enrichment combinations listed in the table were compared, for the entire neutron and photon energy spectrum used in the dose analyses. The comparison shows, that in all cases, the source terms for the higher burnup, with the correspondingly listed enrichment, bound those with any lower burnup in the same burnup column, again with the correspondingly listed enrichment. The burnup column refers to either the colum of the upper bound or lower bound burnups of the burnup ranges. For example, BWR fuel with 45,000 MWD/MTU and 3.0 wt% enrichment bounds BWR fuel with 40,000 MWD/MTU and 2.9 wt% enrichment, or PWR fuel with 50,000 MWD/MTU and 3.9 wt% bounds PWR fuel with 45,000 MWD/MTU and 3.6 wt% enrichment. Also, the maximum burnup for PWR and BWR, together with the correspondingly listed enrichment, bounds all other burnup and enrichment combination for PWR and BWR fuel, respectively. Using this comparison, it may be possible to bound the entire inventory with a single burnup and enrichment value (in combination with a lower bound cooling time). Additional considerations for this approach are as follows:

Table 5.2.36

DESCRIPTION OF FUEL ASSEMBLY USED TO ANNALYZE THORIA RODS IN THE THORIA ROD CANISTER

	BWR
Fuel type	8x8
Active fuel length (in.)	110.5
No. of UO_2 fuel rods	55
No. of UO_2/ThO_2 fuel rods	9
Rod pitch (in.)	0.523
Cladding material	Zircaloy
Rod diameter (in.)	0.412
Cladding thickness (in.)	0.025
Pellet diameter (in.)	0.358
Pellet material	98.2% ThO_2 and 1.8% UO_2 for UO_2/ThO_2 rods or 98.5% ThO_2 and 1.5% UO_2 for UO_2/ThO_2 rods
Pellet density (gm/cc)	10.412
Enrichment (w/o ^{235}U)	93.5 in UO_2 for UO_2/ThO_2 rods and 1.8 for UO_2 rods
Burnup (MWD/MTIHM)	16,000
Cooling Time (years)	18
Specific power (MW/MTIHM)	16.5
Weight of THO_2 and UO_2 (kg)†	121.46
Weight of U (kg)†	92.29
Weight of Th (kg)†	14.74

† Derived from parameters in this table.

Table 5.2.37

CALCULATED FUEL GAMMA SOURCE FOR THORIA ROD CANISTER CONTAINING EIGHTEEN THORIA RODS

98.2% ThO_2 and 1.8% UO_2 for UO_2/ThO_2 rods

Lower Energy	**Upper Energy**	**16,000 MWD/MTIHM 18-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	3.07e+13	5.34e+13
7.0e-01	1.0	5.79e+11	6.81e+11
1.0	1.5	3.79e+11	3.03e+11
1.5	2.0	4.25e+10	2.43e+10
2.0	2.5	4.16e+8	1.85e+8
2.5	3.0	2.31e+11	8.39e+10
Totals		~~1.23e+12~~3.20e+13	~~1.09e+12~~5.45e+13

98.5% ThO_2 and 1.5% UO_2 for UO_2/ThO_2 rods

Lower Energy	**Upper Energy**	**16,000 MWD/MTIHM 18-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	2.88e+13	5.02e+13
7.0e-01	1.0	5.38e+11	6.33e+11
1.0	1.5	3.48e+11	2.79e+11
1.5	2.0	4.04e+10	2.31e+10
2.0	2.5	3.92e+08	1.74e+08
2.5	3.0	2.39e+11	8.67e+10
Totals		3.00e+13	5.12e+13

Table 5.2.38

CALCULATED FUEL NEUTRON SOURCE FOR THORIA ROD CANISTER CONTAINING EIGHTEEN THORIA RODS

98.2% ThO_2 and 1.8% UO_2 for UO_2/ThO_2 rods

Lower Energy (MeV)	Upper Energy (MeV)	16,000 MWD/MTIHM 18-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	5.65e+2
4.0e-01	9.0e-01	3.19e+3
9.0e-01	1.4	6.79e+3
1.4	1.85	1.05e+4
1.85	3.0	3.68e+4
3.0	6.43	1.41e+4
6.43	20.0	1.60e+2
Totals		7.21e+4

98.5% ThO_2 and 1.5% UO_2 for UO_2/ThO_2 rods

Lower Energy (MeV)	Upper Energy (MeV)	16,000 MWD/MTIHM 18-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	5.99e+2
4.0e-01	9.0e-01	3.39e+3
9.0e-01	1.4	7.21e+3
1.4	1.85	1.11e+4
1.85	3.0	3.91e+4
3.0	6.43	1.50e+4
6.43	20.0	1.69e+2
Totals		7.66e+4

of antimony. A larger mass of antimony is conservative since the calculated activity of Sb-124 is directly proportional to the initial mass of antimony.

The number of gammas from fuel assemblies with energies greater than 1.666 MeV entering the 77.25 inch long neutron source was calculated to be 1.04E+8 gammas/sec which would produce a neutron source of 603.2 neutrons/sec (1.04E+8 * 5.8E-6). The steady state amount of Sb-124 activated in the antimony was calculated to be 39.9 curies. This activity level would produce a neutron source of 4.63E+6 neutrons/sec (39.9 * 1.16E+5) or 6.0E+4 neutrons/sec/inch (4.63E+6/77.25). These calculations conservatively neglect the reduction in antimony and beryllium which would have occurred while the neutron sources were in the core and being irradiated at full reactor power.

Since this is a localized source (77.25 inches in length) it is appropriate to compare the neutron source per inch from the design basis Dresden Unit 1 fuel assembly, 6x6, containing an Sb-Be neutron source to the design basis fuel neutron source per inch. This comparison, presented in Table 5.4.18, demonstrates that a Dresden Unit 1 fuel assembly containing an Sb-Be neutron source is bounded by the design basis fuel.

As stated above, the Sb-Be source is encased in a steel rod. Therefore, the gamma source from the activation of the steel was considered assuming a burnup of 120,000 MWD/MTU which is the maximum burnup assuming the Sb-Be source was in the reactor for the entire 18 year life of Dresden Unit 1. The cooling time assumed was 18 years which is the minimum cooling time for Dresden Unit 1 fuel. The source from the steel was bounded by the design basis fuel assembly. In conclusion, storage of a Dresden Unit 1 Sb-Be neutron source in a Dresden Unit 1 fuel assembly is acceptable and bounded by the current analysis.

5.4.8 Thoria Rod Canister

Based on a comparison of the gamma spectra from Tables 5.2.37 and 5.2.7 for the thoria rod canister and design basis 6x6 fuel assembly, respectively, it is difficult to determine if the thoria rods will be bounded by the 6x6 fuel assemblies. However, it is obvious that the neutron spectra from the 6x6, Table 5.2.18, bounds the thoria rod neutron spectra, Table 5.2.38, with a significant margin. In order to demonstrate that the gamma spectrum from the single thoria rod canister is bounded by the gamma spectrum from the design basis 6x6 fuel assembly, the gamma dose rate on the outer radial surface of the 100-ton HI-TRAC and the HI-STORM overpack was estimated conservatively assuming an MPC full of thoria rod canisters. This gamma dose rate was compared to an estimate of the dose rate from an MPC full of design basis 6x6 fuel assemblies. The gamma dose rate from the 6x6 fuel was higher for the 100-ton HI-TRAC and only ~2015% lower for the HI-STORM overpack than the dose rate from an MPC full of thoria rod canisters. This in conjunction with the significant margin in neutron spectrum and the fact that there is only one thoria rod canister clearly demonstrates that the thoria rod canister is acceptable for storage in the MPC-68 or the MPC-68F.

Per Supplement 5.III, the effect of the design differences between the MPC-68 and MPC-68M on the dose rates is small and all results and conclusions from the MPC-68 are applicable to the MPC-68M. Therefore, the thoria rod canister is also acceptable for storage in the MPC-68M.

5.4.9 Regionalized Loading Dose Rate Evaluation

Section 2.1.9 describes the regionalized loading scheme available in the HI-STORM 100 system. Depending on the choice of X (the ratio of inner region assembly heat load to outer region assembly heat load), higher heat load fuel (higher burnup and shorter cooling time) may be placed in either region 1 or region 2. If X is greater than 1, the higher heat load fuel is placed in region 1 and shielded by lower heat load fuel in region 2. This configuration produces the lowest dose rates since the older colder fuel is being used as shielding for the younger hotter fuel. If X is less than 1, then the younger hotter fuel is placed on the periphery of the basket and the older colder fuel is placed on the interior of the basket. This configuration will result in higher radial dose rates than for configurations with X greater than or equal to 1. In order to perform a bounding shielding analysis, the burnup and cooling time combinations listed in Section 5.1 were chosen to bound all values of X. All fuel assemblies in an MPC were assumed to have the same burnup and cooling time in the shielding analysis. This approach results in dose rates calculated in this chapter that bound all allowable regionalized and uniform loading burnup and cooling time combinations.

5.4.10 Fuel Assemblies with Stainless Steel Replacement Rods Dose Rate Evaluation

A dose rate evaluation for the HI-STORM 100S Version B containing the MPC-32 and the MPC-68 is performed to determine the impact of storing fuel assemblies with irradiated stainless steel replacement rods. The stainless steel rods are irradiated in the same neutron flux and for the same time period as the design basis PWR and BWR UO2 fuel rods. The dose rates at several locations, adjacent to and at 1 meter, from the HI-STORM containing the MPC-32 are presented in Table 5.1.11 and Table 5.1.14, respectively. The dose rates for the HI-STORM containing the MPC-68 are presented in Tables 5.1.13 and Table 5.1.16. The dose rates at the same locations are calculated assuming all 32 design basis PWR assemblies contain 4 irradiated stainless steel replacement rods and all 68 design basis BWR assemblies contain 2 irradiated stainless steel replacement rods. The dose rates with the 4 irradiated stainless steel replacement rods in the design basis PWR assembly are approximately 10% higher at the sides and top of the HI-STORM containing the MPC-32. The dose rates with the 2 irradiated stainless steel replacement rods in the design basis BWR assembly are approximately 33% higher at the sides and top of the HI-STORM containing the MPC-68. Therefore, fuel assemblies containing irradiated stainless steel replacement rods are acceptable for storage and, if present in a fuel assembly, need to be considered in the site specific dose calculations.

APPENDIX 5.F

[INTENTIONALLY DELETED]

~~Additional Information on the Burnup Versus Decay Heat and Enrichment Equation~~

The equation in Section 5.2.5.3 was determined to be the best equation capable of reproducing the burnup versus enrichment and decay heat data calculated with ORIGEN-S. As an example, Figure 5.F.1 graphically presents ORIGEN-S burnup versus decay heat data for various enrichments for the 9x9C/D fuel assembly array/classes with a 20- year cooling time. This data could also be represented graphically as a surface on a three dimensional plot. However, the 2D plot is easier to visualize. Additional enrichments were used in the ORIGEN-S calculations and have been omitted for clarity.

Figures 5.F.2 through 5.F.4 show ORIGEN-S burnup versus decay heat data for specific enrichments. In addition to the ORIGEN-S data, these figures present the results of the original curve fit and the adjusted curve fit. Table 5.F.1 below shows the equation coefficients used for both curve fits. As these figures indicate, the curve fit faithfully reproduces the ORIGEN-S data.

Figure 5.F.5 provides a different representation of the curve fit versus ORIGEN-S comparison. This figure was generated by taking the ORIGEN-S enrichment and decay heat data from Figure 5.F.1 for a constant burnup of 30,000 MWD/MTU and calculating the burnup using the fitted equation with coefficients from Table 5.F.1. The resulting burnup versus enrichment is plotted. Table 5.F.2 presents the ORIGEN-S and curve fit data in tabular form used to generate Figure 5.F.5. Since the ORIGEN-S calculations were performed for a specific burnup of 30,000 MWD/MTU, the ORIGEN-S data is represented as a straight line. Figures 5.F.6 and 5.F.7 provide the same representation for burnups of 45,000 and 65,000 MWD/MTU. These results also indicate that the non-adjusted curve fit provides a very good representation of the ORIGEN-S data. It is also clear that the adjusted curve fit always bounds the ORIGEN-S data by predicting a lower burnup which results in a more restrictive and conservative limit for the user.

Table 5.F.1

[INTENTIONALLY DELETED]

~~COEFFICIENTS FOR EQUATION IN SECTION 5.2.5.3 FOR THE 9X9C/D FUEL ASSEMBLY ARRAY/CLASSES WITH A COOLING TIME OF 20 YEARS~~

~~Coefficient~~	~~Original Curve Fit~~	~~Adjusted Curve Fit~~
~~A~~	~~249944~~	~~249944~~
~~B~~	~~-382059~~	~~-382059~~
~~C~~	~~308281~~	~~308281~~
~~D~~	~~-205.495~~	~~-205.495~~
~~E~~	~~9362.63~~	~~9362.63~~
~~F~~	~~1389.71~~	~~1389.71~~
~~G~~	~~-1995.54~~	~~-2350.49~~

Table 5.F.2

[INTENTIONALLY DELETED]

ORIGEN-S AND CURVE FIT DATA FOR THE 9X9C/D FUEL ASSEMBLY ARRAY/CLASSES
WITH A COOLING TIME OF 20 YEARS

Specified Enrichment	ORIGEN-S calculated decay heat per assembly (kw)	ORIGEN-S calculated burnup (MWD/MTU)	Burnup calculated with original curve fit (MWD/MTU)	Burnup calculated with adjusted curve fit (MWD/MTU)
0.7	1.55E-01	30000	29700.69	29345.74
1	1.53E-01	30000	29715.24	29360.29
1.35	1.52E-01	30000	29759.8	29404.85
1.7	1.50E-01	30000	29849.09	29494.14
2	1.50E-01	30000	29997.43	29642.48
2.3	1.49E-01	30000	30050.56	29695.61
2.6	1.49E-01	30000	30120.16	29765.21
2.9	1.49E-01	30000	30228.56	29873.61
3.2	1.50E-01	30000	30340.01	29985.06
3.4	1.50E-01	30000	30354.95	30000
3.6	1.49E-01	30000	30172.21	29817.26
3.9	1.48E-01	30000	30095.41	29740.46
4.2	1.48E-01	30000	30001.17	29646.22
4.5	1.48E-01	30000	29890.42	29535.47
4.8	1.48E-01	30000	29764.09	29409.14
5	1.49E-01	30000	29731.66	29376.71

[INTENTIONALLY DELETED]

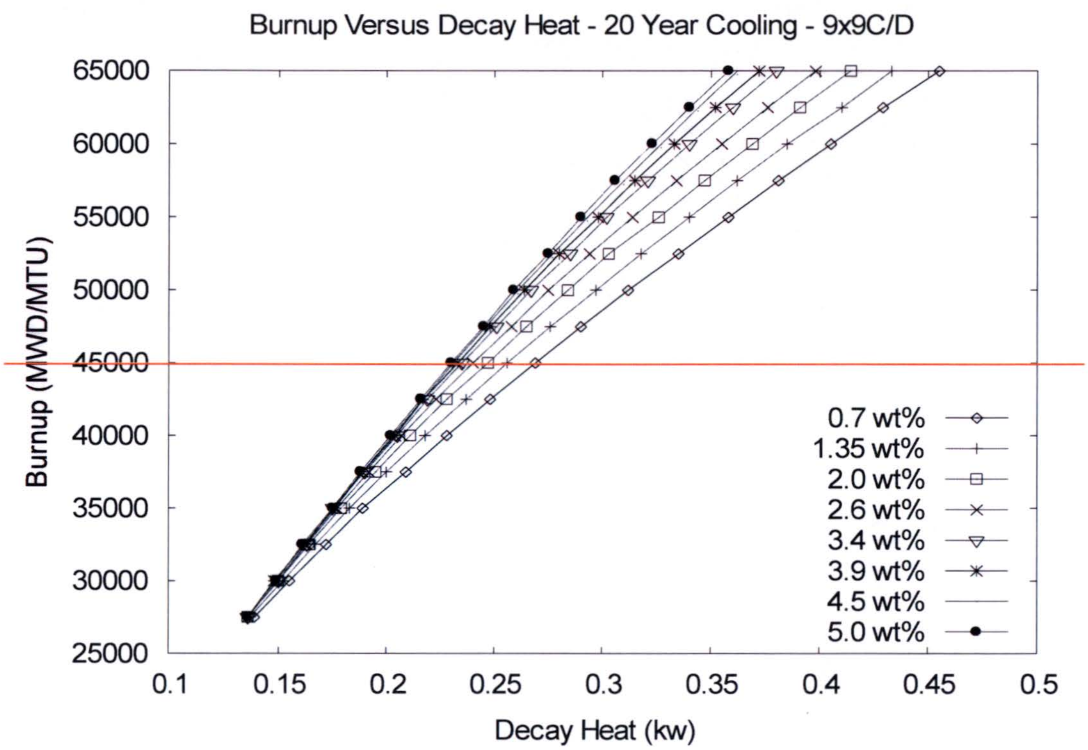

FIGURE 5.F.1; ~~ORIGEN-S CALCULATED BURNUP VERSUS DECAY HEAT FOR VARIOUS ENRICHMENTS~~

[INTENTIONALLY DELETED]

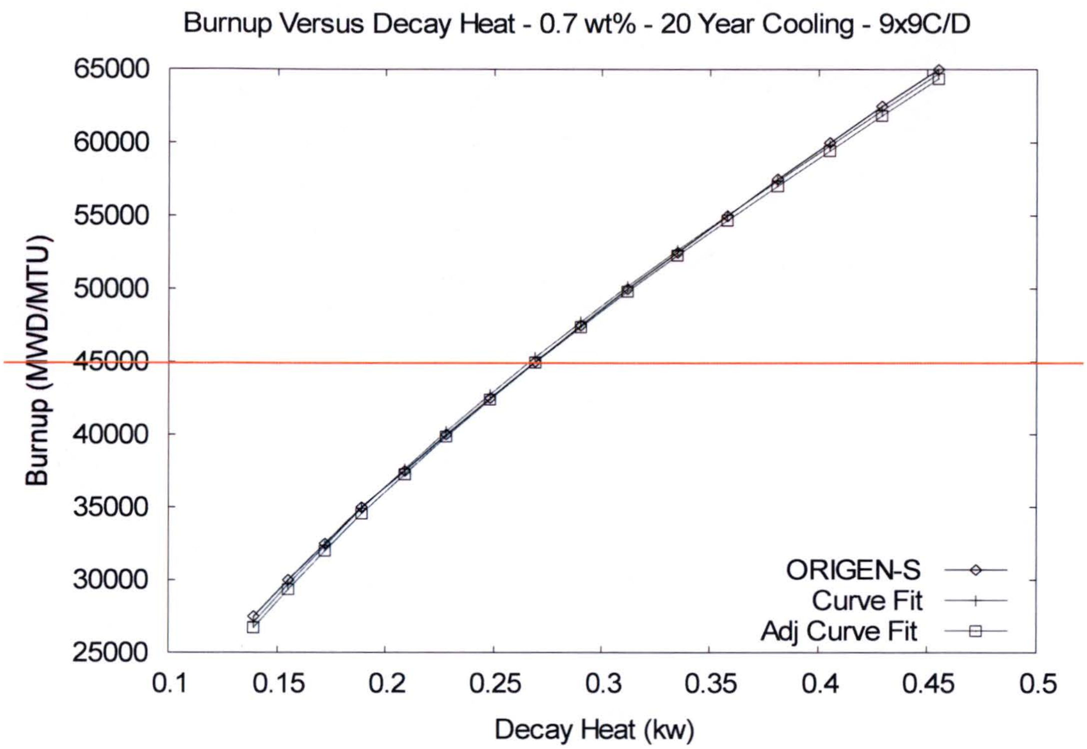

FIGURE 5.F.2; ~~A COMPARISON OF THE BURNUP VERSUS DECAY HEAT CALCULATIONS FROM ORIGEN-S, THE ORIGINAL CURVE FIT, AND THE ADJUSTED CURVE FIT FOR AN ENRICHMENT OF 0.7 WT.% ^{235}U.~~

[INTENTIONALLY DELETED]

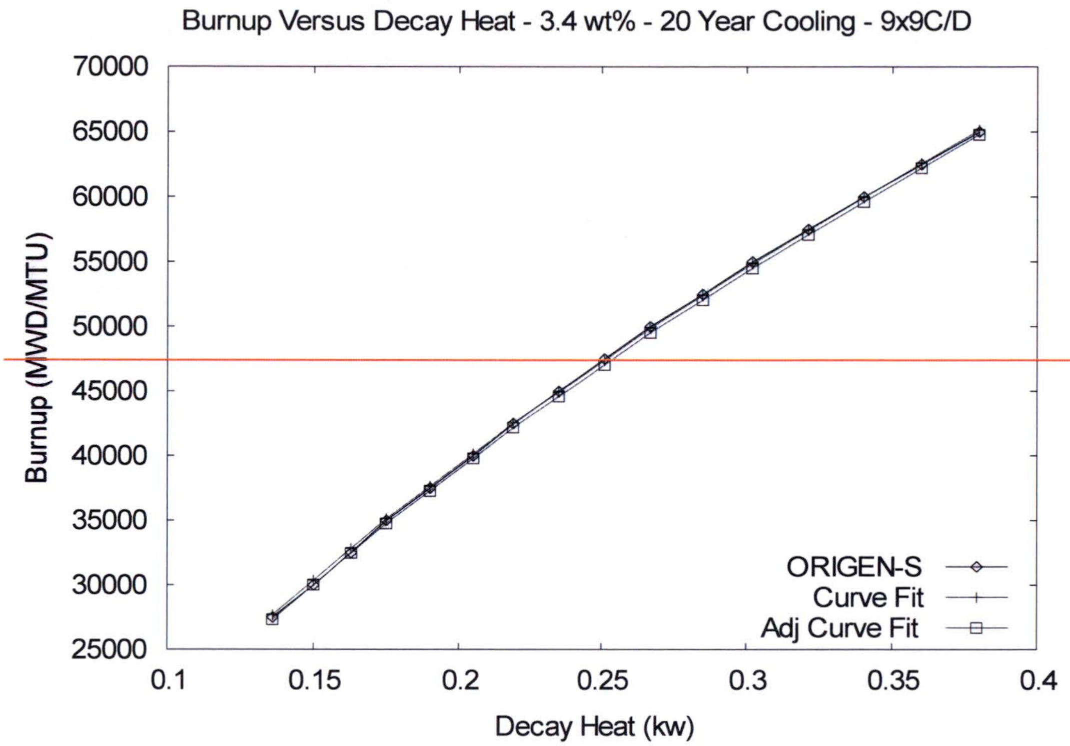

FIGURE 5.F.3; ~~A COMPARISON OF THE BURNUP VERSUS DECAY HEAT CALCULATIONS FROM ORIGEN-S, THE ORIGINAL CURVE FIT, AND THE ADJUSTED CURVE FIT FOR AN ENRICHMENT OF 3.4 WT.% ^{235}U.~~

[INTENTIONALLY DELETED]

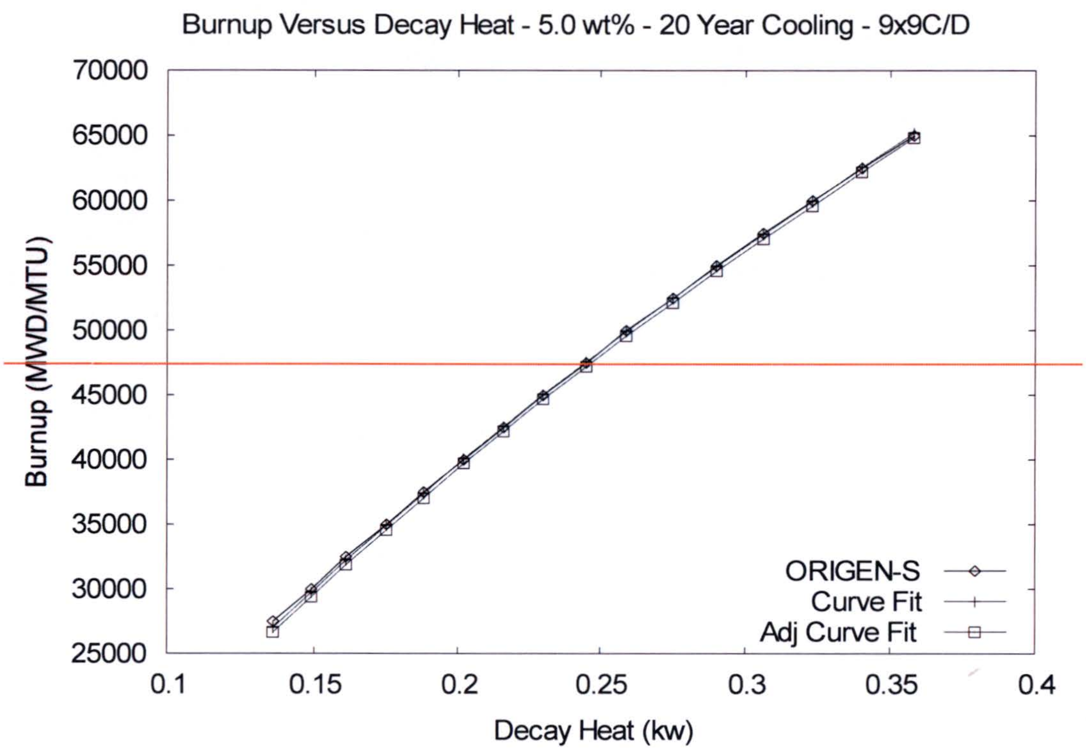

FIGURE 5.F.4; ~~A COMPARISON OF THE BURNUP VERSUS DECAY HEAT CALCULATIONS FROM ORIGEN-S, THE ORIGINAL CURVE FIT, AND THE ADJUSTED CURVE FIT FOR AN ENRICHMENT OF 5.0 WT.% ^{235}U.~~

[INTENTIONALLY DELETED]

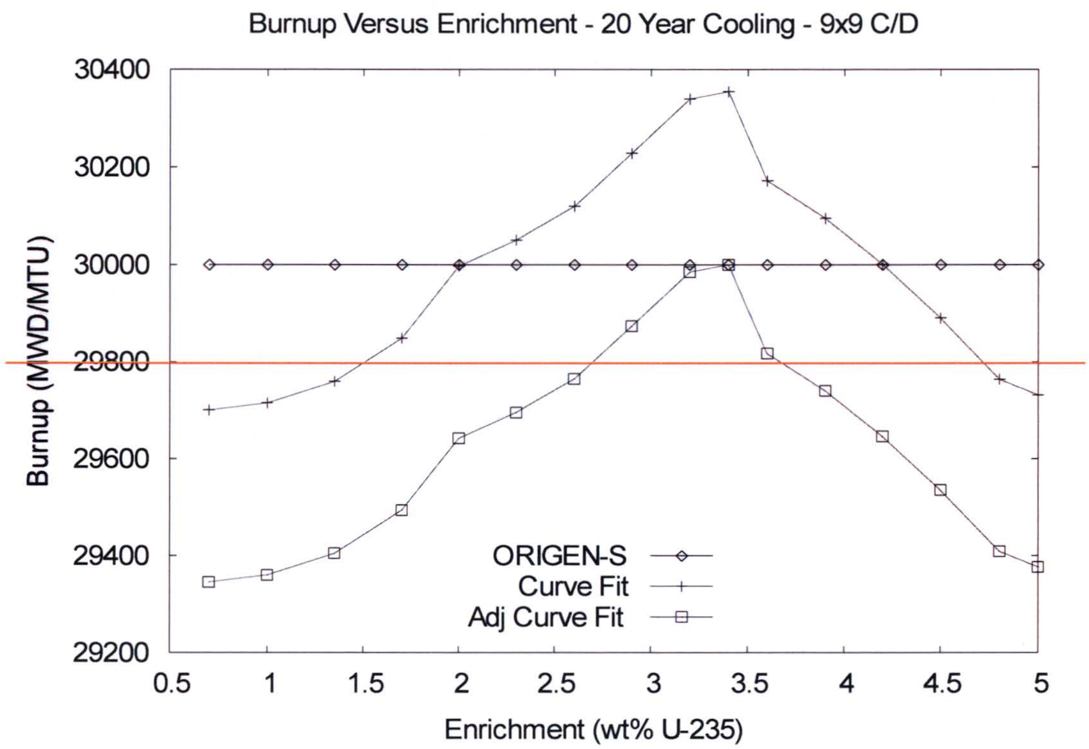

FIGURE 5.F.5; A COMPARISON OF THE CALCULATED BURNUPS USING THE CURVE FIT AND THE ADJUSTED CURVE FIT FOR VARIOUS ENRICHMENTS. ALL ORIGEN-S CALCULATIONS YIELDED A BURNUP OF 30,000 MWD/MTU.

[INTENTIONALLY DELETED]

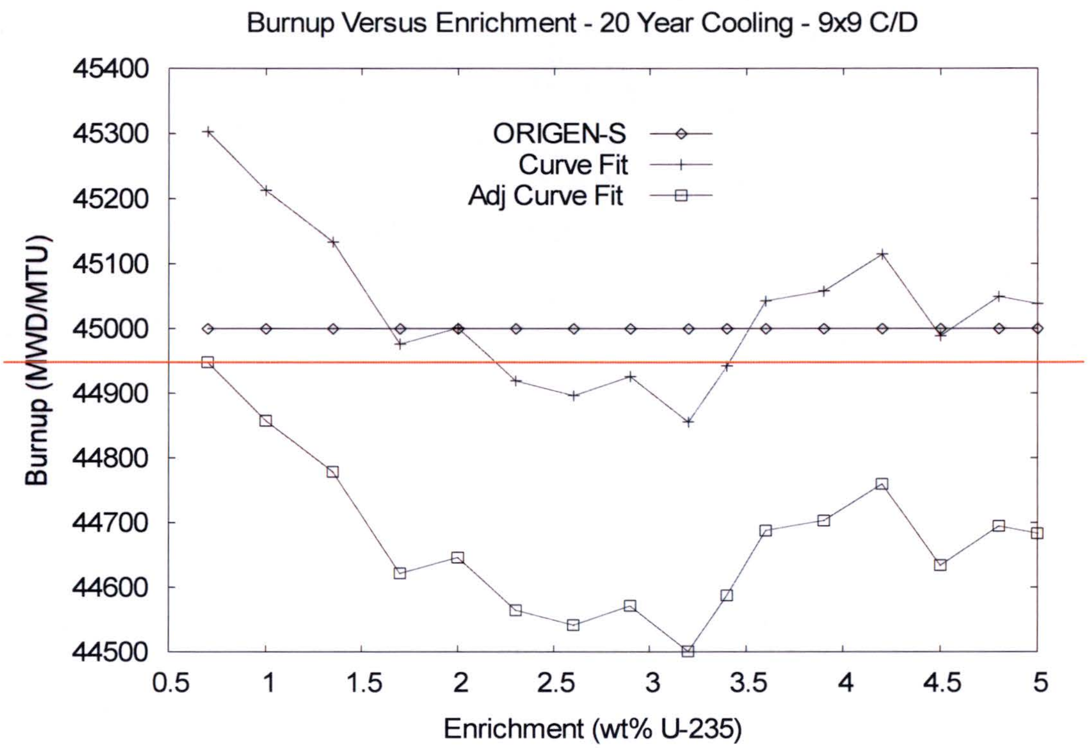

FIGURE 5.F.6; ~~A COMPARISON OF THE CALCULATED BURNUPS USING THE CURVE FIT AND THE ADJUSTED CURVE FIT FOR VARIOUS ENRICHMENTS. ALL ORIGEN-S CALCULATIONS YIELDED A BURNUP OF 45,000 MWD/MTU.~~

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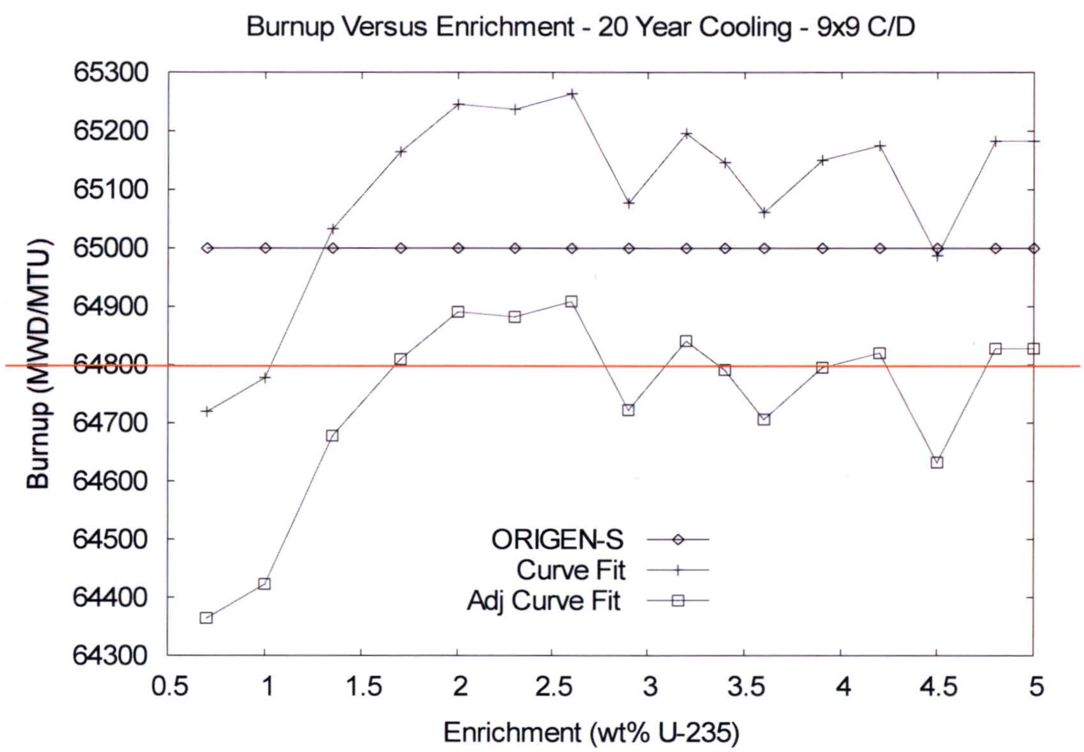

FIGURE 5.F.7; ~~A COMPARISON OF THE CALCULATED BURNUPS USING THE CURVE FIT AND THE ADJUSTED CURVE FIT FOR VARIOUS ENRICHMENTS. ALL ORIGEN-S CALCULATIONS YIELDED A BURNUP OF 65,000 MWD/MTU.~~

conservatively truncated to 1.0000, consistent with NUREG-1536.

- 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 further discussions see Section 6.3.3.

- For intact fuel assemblies, as defined in Table 1.0.1, 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. The number of dummy rods used to replace missing fuel rods is not limited.

Results of the design basis criticality safety calculations for single internally flooded HI-TRAC transfer casks with full water reflection on all sides (limiting cases for the HI-STORM 100 System), and for single unreflected, internally flooded HI-STAR casks (limiting cases for the HI-STAR 100 System), loaded with intact fuel assemblies are listed in Tables 6.1.1 through 6.1.8, conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.3), and including the calculational bias, uncertainties, and calculational statistics. Comparing corresponding results for the HI-TRAC and HI-STAR demonstrates that the overpack material does not significantly affect the reactivity. Consequently, analyses for the HI-STAR System are directly applicable to the HI-STORM 100 System and vice versa. In addition, a few results for single internally dry (no moderator) HI-STORM storage casks with full water reflection on all external surfaces of the overpack, including the annulus region between the MPC and overpack, are listed to confirm the low reactivity of the HI-STORM 100 System in storage.

For each of the MPC designs, minimum soluble boron concentration (if applicable) and fuel assembly classes[††], Tables 6.1.1 through 6.1.8 list the bounding maximum k_{eff} value, and the associated maximum allowable enrichment. The maximum allowed enrichments and the minimum soluble boron concentrations are also listed in Section 2.1.9. The candidate fuel assemblies, that are bounded by those listed in Tables 6.1.1 through 6.1.8, are given in Section 6.2.

[††] For each array size (e.g., 6x6, 7x7, 14x14, etc.), the fuel assemblies have been subdivided into a number of assembly classes, where an assembly class is defined in terms of the (1) number of fuel rods; (2) pitch; (3) number and location of guide tubes (PWR) or water rods (BWR); and (4) cladding material. The assembly classes for BWR and PWR fuel are defined in Section 6.2.

Table 6.1.5

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-32 FOR 4.1% ENRICHMENT

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Minimum Soluble Boron Concentration (ppm)*	Maximum† k_{eff}		
			HI-STORM	HI-TRAC	HI-STAR
14x14A	4.1	1300	---	---	0.9041
14x14B	4.1	1300	---	---	0.9257
14x14C	4.1	1300	---	---	0.9423
14x14D	4.1	1300	---	---	0.8970
14x14E	4.1	1300	---	---	0.7340
15x15A	4.1	1800	---	---	0.9206
15x15B	4.1	1800	---	---	0.9397
15x15C	4.1	1800	---	---	0.9266
15x15D	4.1	1900	---	---	0.9384
15x15E	4.1	1900	---	---	0.9365
15x15F	4.1	1900	0.4691	0.9403	0.9411
15x15G	4.1	1800	---	---	0.9147
15x15H	4.1	1900	---	---	0.9276
15x15I	4.1	1800	---	---	0.9340
16x16A	4.1	1400	---	---	0.9375
16x16B	4.1	1400	---	---	0.9354
16x16C	4.1	1400	---	---	0.9178
17x17A	4.1	~~1900~~1600	---	---	~~0.9111~~0.9421
17x17B	4.1	1900	---	---	0.9309
17x17C	4.1	1900	---	0.9365	0.9355

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

* For maximum allowable enrichments between 4.1 wt% ^{235}U and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified in Table 6.1.5 and Table 6.1.6 for each assembly class.

† The term "maximum k_{eff}" as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.1.6

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-32 FOR 5.0% ENRICHMENT

Fuel Assembly Class	**Maximum Allowable Enrichment (wt% ^{235}U)**	**Minimum Soluble Boron Concentration (ppm)** *	**Maximum† k_{eff}**		
			HI-STORM	**HI-TRAC**	**HI-STAR**
14x14A	5.0	1900	---	---	0.9000
14x14B	5.0	1900	---	---	0.9214
14x14C	5.0	1900	---	---	0.9480
14x14D	5.0	1900	---	---	0.9050
14x14E	5.0	1900	---	---	0.7415
15x15A	5.0	2500	---	---	0.9230
15x15B	5.0	2500	---	---	0.9429
15x15C	5.0	2500	---	---	0.9307
15x15D	5.0	2600	---	---	0.9466
15x15E	5.0	2600	---	---	0.9434
15x15F	5.0	2600	0.5142	0.9470	0.9483
15x15G	5.0	2500	---	---	0.9251
15x15H	5.0	2600	---	---	0.9333
15x15I	5.0	2500	---	---	0.9402
16x16A	5.0	2000	---	---	0.9429
16x16B	5.0	2000	---	---	0.9378
16x16C	5.0	2000	---	---	0.9208
17x17A	5.0	~~2600~~2200	---	---	~~0.9161~~0.9475
17x17B	5.0	2600	---	---	0.9371
17x17C	5.0	2600	---	0.9436	0.9437

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

* For maximum allowable enrichments between 4.1 wt% ^{235}U and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified in Table 6.1.5 and Table 6.1.6 for each assembly class.

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.1.12

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-32
WITH UP TO 8 DFCs

Fuel Assembly Class of Intact Fuel	**Maximum Allowable Enrichment for Intact Fuel and Damaged Fuel/Fuel Debris (wt% ^{235}U)**	**Minimum Soluble Boron Content (ppm)** †	**Maximum k_{eff}**	
			HI-TRAC	**HI-STAR**
14x14A, B, C, D, E	4.1	1500	---	0.9336
	5.0	2300	---	0.9269
15x15A, B, C, G, I	4.1	1900	0.9349	0.9350
	5.0	2700	---	0.9365
15x15D, E, F, H	4.1	2100	---	0.9340
	5.0	2900	0.9382	0.9397
16x16A, B, C	4.1	1500	---	0.9348
	5.0	2300	---	0.9299
17x17A	4.1	1800	---	0.9311
	5.0	2600	---	0.9298
17x17~~A,~~ B, C	4.1	2100	---	0.9294
	5.0	2900	---	0.9367

† For maximum allowable enrichments between 4.1 wt% ^{235}U and 5.0 wt% ^{235}U, the minimum soluble boron concentration may be calculated by linear interpolation between the minimum soluble boron concentrations specified for each assembly class.

fuel debris. The requirements for damaged fuel and fuel debris in the MPC-24E is discussed in Section 6.2.4.3.

Without credit for soluble boron, the maximum allowable fuel enrichment varies between 4.2 and 5.0 wt% ^{235}U, depending on the assembly classes as identified in Tables 6.2.6 through 6.2.22. The maximum allowable enrichment for each assembly class is listed in Table 6.1.3, together with the maximum k_{eff} for the bounding assembly in the assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit The 15x15F assembly class at 4.5% enrichment has the highest reactivity (maximum k_{eff} shown in Table 6.1.3). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

For a minimum soluble boron concentration of 300ppm, the maximum allowable fuel enrichment is 5.0 wt% ^{235}U for all assembly classes identified in Tables 6.2.6 through 6.2.22. Table 6.1.4 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit. The 15x15H assembly class has the highest reactivity (maximum k_{eff} shown in Table 6.1.4). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

6.2.2.4 Intact PWR Assemblies in the MPC-32

When loading any PWR fuel assembly in the MPC-32, a minimum soluble boron concentration is required.

For a maximum allowable fuel enrichment of 4.1 wt% ^{235}U for all assembly classes identified in Tables 6.2.6 through 6.2.22, a minimum soluble boron concentration between 1300ppm and 1900ppm is required, depending on the assembly class. Table 6.1.5 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit. The 15x15F 17x17A assembly class has the highest reactivity (maximum k_{eff} shown in Table 6.1.5). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

For a maximum allowable fuel enrichment of 5.0 wt% ^{235}U for all assembly classes identified in Tables 6.2.6 through 6.2.22, a minimum soluble boron concentration between 1900ppm and 2600ppm is required, depending on the assembly class. Table 6.1.6 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit. The 15x15F assembly class has the highest reactivity (maximum k_{eff} shown in Table 6.1.6). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

It is desirable to limit the soluble boron concentration to a level appropriate for the maximum enrichment in a basket, since this prevents adding soluble boron unnecessarily to the spent fuel pool during loading and unloading operations. This approach requires a minimum soluble boron

6.2.4.3 Damaged PWR Fuel Assemblies and Fuel Debris

In addition to storing intact PWR fuel assemblies, the HI-STORM 100 System is designed to store damaged PWR fuel assemblies and fuel debris (MPC-24E and MPC-32). Damaged fuel assemblies and fuel debris are defined in Table 1.0.1. Damaged PWR fuel assemblies and fuel debris are required to be loaded into PWR Damaged Fuel Containers (DFCs).

6.2.4.3.1 Damaged PWR Fuel Assemblies and Fuel Debris in the MPC-24E

Up to four DFCs may be stored in the MPC-24E. When loaded with damaged fuel and/or fuel debris, the maximum enrichment for intact and damaged fuel is 4.0 wt% ^{235}U for all assembly classes listed in Table 6.2.6 through 6.2.22 without credit for soluble boron. The maximum k_{eff} for these classes is 0.9486. For a minimum soluble boron concentration of 600ppm, the maximum enrichment for intact and damaged fuel is 5.0 wt% ^{235}U for all assembly classes listed in Table 6.2.6 through 6.2.22. The criticality evaluation of the damaged fuel is presented in Subsection 6.4.4.2.

6.2.4.3.2 Damaged PWR Fuel Assemblies and Fuel Debris in the MPC-32

Up to eight DFCs may be stored in the MPC-32. For a maximum allowable fuel enrichment of 4.1 wt% ^{235}U for intact fuel, damaged fuel and fuel debris for all assembly classes identified in Tables 6.2.6 through 6.2.22, a minimum soluble boron concentration between 1500ppm and 2100ppm is required, depending on the assembly class of the intact assembly. For a maximum allowable fuel enrichment of 5.0 wt% ^{235}U for intact fuel, damaged fuel and fuel debris, a minimum soluble boron concentration between 2300ppm and 2900ppm is required, depending on the assembly class of the intact assembly. Table 6.1.12 shows the maximum k_{eff} by assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit.

As discussed in Section 6.2.2.4, it is desirable to limit the soluble boron concentration to a level appropriate for the maximum enrichment in a basket. The discussion presented in Section 6.2.2.4 is also applicable for the MPC-32 with damaged fuel or fuel debris. Further, studies with damaged fuel have shown that this approach also results in maximum k_{eff} values that are lower than those k_{eff} values calculated for 4.1 wt% and 5.0 wt% ^{235}U in Table 6.1.12.

6.2.5 Thoria Rod Canister

Additionally, the HI-STORM 100 System is designed to store a Thoria Rod Canister in the MPC-68 or MPC-68F. The canister is similar to a DFC and contains 18 intact Thoria Rods placed in a separator assembly. The reactivity of the canister in the MPC is very low compared to the approved fuel assemblies (The ^{235}U content of these rods correspond to UO_2 rods with an initial enrichment of approximately up to 1.7 wt% ^{235}U). It is therefore permissible to the Thoria Rod Canister together with any approved content in a MPC-68 or MPC-68F. Specifications of the canister and the Thoria Rods that are used in the criticality evaluation are given in Table

Table 6.2.46

SPECIFICATION OF THE THORIA ROD CANISTER AND THE THORIA RODS

Canister ID	4.81"
Canister Wall Thickness	0.11"
Separator Assembly Plates Thickness	0.11"
Cladding OD	0.412"
Cladding ID	0.362"
Pellet OD	0.358"
Active Length	110.5"
Fuel Composition	1.8% UO_2 and 98.2% ThO_2 or 1.5% UO_2 and 98.5% ThO_2
Initial Enrichment	93.5 wt% ^{235}U for 1.8% of the fuel
Maximum k_{eff}†	0.1813
Calculated k_{eff}	0.1779
Standard Deviation	0.0004

† The maximum calculated k_{eff} of 0.1813 assumes an average ThO_2 content of 98.2 wt%. It is also based on a UO_2 content of 1.8 wt%. Reducing the UO_2 content from 1.8 wt% to the average value of 1.5 wt% would result in a reduction of the already low reactivity, due to the reduction in the fissile material. Therefore the values listed in the table are bounding.

Table 6.3.2 (cont.)

MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES[†]

Pitch	Box I.D.	Box Wall Thickness	MCNP4a Calculated k_{eff}
MPC-24 (17x17A @ 5.0% Enrichment) 400ppm soluble boron			
nominal (10.906")	maximum (8.98")	nominal (5/16")	0.9236±0.0007[††]
maximum (10.966")	maximum (8.98")	nominal (5/16")	0.9176±0.0008
minimum (10.846")	nominal (8.92")	nominal (5/16")	0.9227±0.0010
minimum (10.846")	minimum (8.86")	nominal (5/16")	0.9159±0.0008
nominal (10.906")	nominal-0.04" (8.88")	nom.+0.05" (0.3625")	0.9232±0.0009
nominal (10.906")	nominal (8.92")	nominal (5/16")	0.9158±0.0007
MPC-32 (17x17A @ 5.0% Enrichment) ~~2600~~ 2200 ppm soluble boron[†††]			
minimum (9.158")	minimum (8.69")	nominal (9/32")	0.9399±0.0007 ~~0.9085±0.0007~~
nominal (9.218")	nominal (8.75")	nominal (9/32")	0.9370±0.0007 ~~0.9028±0.0007~~
maximum (9.278")	maximum (8.81")	nominal (9/32")	0.9313±0.0008 ~~0.8996±0.0008~~
nominal+0.05" (9.268")	nominal (8.75")	nominal+0.05" (0.331")	0.9356±0.0007 ~~0.9023±0.0008~~
minimum+0.05"(9.208")	minimum (8.69")	nominal+0.05" (0.331")	0.9395±0.0008 ~~0.9065±0.0007~~
maximum (9.278")	Maximum-0.05" (8.76")	nominal+0.05" (0.331")	0.9330±0.0008 ~~0.9030±0.0008~~

Notes:

1. Values in parentheses are the actual value used.

† Tolerance for pitch and box I.D. are ± 0.06". Tolerance for box wall thickness is +0.05", -0.00".

†† Numbers are 1σ statistical uncertainties.

††† for 0.075" sheathing thickness. See Section 6.3.1 and Table 6.3.5 for reactivity effect of sheathing thickness.

6.4.4.2.1 Bounding Intact Assemblies

Intact BWR assemblies stored together with DFCs are limited to a maximum planar average enrichment of 3.7 wt% ^{235}U, regardless of the fuel class. The results presented in Table 6.1.7 are for different enrichments for each class, ranging between 2.7 and 4.2 wt% ^{235}U, making it difficult to identify the bounding assembly. Therefore, additional calculations were performed for the bounding assembly in each assembly class with a planar average enrichment of 3.7 wt%. The results are summarized in Table 6.4.7 and demonstrate that the assembly classes 9x9E and 9x9F have the highest reactivity. These two classes share the same bounding assembly (see footnotes for Tables 6.2.33 and 6.2.34 for further details). This bounding assembly is used as the intact BWR assembly for all calculations with DFCs.

Intact PWR assemblies stored together with DFCs in the MPC-24E are limited to a maximum enrichment of 4.0 wt% ^{235}U without credit for soluble boron and to a maximum enrichment of 5.0 wt% with credit for soluble boron, regardless of the fuel class. The results presented in Table 6.1.3 are for different enrichments for each class, ranging between 4.2 and 5.0 wt% ^{235}U, making it difficult to directly identify the bounding assembly. However, Table 6.1.4 shows results for an enrichment of 5.0 wt% for all fuel classes, with a soluble boron concentration of 300 ppm. The assembly class 15x15H has the highest reactivity. This is consistent with the results in Table 6.1.3, where the assembly class 15x15H is among the classes with the highest reactivity, but has the lowest initial enrichment. Therefore, in the MPC-24E, the 15x15H assembly is used as the intact PWR assembly for all calculations with DFCs.

Intact PWR assemblies stored together with DFCs in the MPC-32 are limited to a maximum enrichment of 5.0 wt%, regardless of the fuel class. Table 6.1.5 and Table 6.1.6 show results for enrichments of 4.1 wt% and 5.0 wt%, respectively, for all fuel classes. Since different minimum soluble boron concentrations are used for different groups of assembly classes, the assembly class with the highest reactivity in each group is used as the intact assembly for the calculations with DFCs in the MPC-32. These assembly classes are

- 14x14C for all 14x14 assembly classes;
- 15x15B for assembly classes 15x15A, B, C, G and I;
- 15x15F for assembly classes 15x15D, E, F and H;
- 16x16A for assembly classes 16x16A, B and C;
- 17x17A assembly class and
- 17x17C for assembly classes ~~all~~ 17x17B and C ~~assembly classes~~.

voided guide tubes, a water density of 1.0 g/cc in the DFC and MPC, 2900 ppm soluble boron, and an enrichment of 5.0 wt% ^{235}U for the intact and damaged fuel and fuel debris. For this case, results are summarized in Table 6.4.13. For each condition, the table lists the highest maximum k_{eff}, including bias and calculational uncertainties, i.e. the point of optimum moderation. The results show that the fuel pellet diameter in the DFC has an insignificant effect on reactivity, and that reactivity decreases with decreasing water density. The latter demonstrates that replacing all cladding and other structural material with water is conservative even in the presence of soluble boron in the water. Therefore, a typical fuel pellet diameter and a water density of 1.0 in the DFCs are used for all further analyses. Two enrichment levels are analyzed, 4.1 wt% ^{235}U and 5.0 wt% ^{235}U, consistent with the analyses for intact fuel only. In any calculation, the same enrichment is used for the intact fuel and the damaged fuel and fuel debris. For both enrichment levels, analyses are performed with voided and filled guide tubes, each with water densities of 0.93 and 1.0 g/cm^3 in the MPC. In all cases, the water density inside the DFCs is assumed to be 1.0 g/cm^3, since this is the most reactive condition as shown in Table 6.4.13. Results are summarized in Table 6.4.14. For each group of assembly classes, the table shows the soluble boron level and the highest maximum k_{eff} for the various moderation conditions of the intact assembly. The highest maximum k_{eff} is the highest value of any of the hypothetical fuel debris configurations, i.e. various arrays of bare fuel rods. All maximum k_{eff} values are below the 0.95 regulatory limit. Conditions of damaged fuel such as assemblies with missing rods or collapsed assemblies were not analyzed in the MPC-32, since the results in Figure 6.4.14 clearly demonstrate that these conditions are bounded by the hypothetical model for fuel debris based on regular arrays of bare fuel rods.

6.4.5 Fuel Assemblies with Missing Rods

For fuel assemblies that are qualified for damaged fuel storage, missing and/or damaged fuel rods are acceptable. However, for fuel assemblies to meet the limitations of intact fuel assembly storage, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods. The number of dummy rods that is used to replace missing and/or damaged fuel rods is not limited.

6.4.6 Thoria Rod Canister

The Thoria Rod Canister is similar to a DFC with an internal separator assembly containing 18 intact fuel rods. The configuration is illustrated in Figure 6.4.15. The k_{eff} value for an MPC-68F filled with Thoria Rod Canisters is calculated to be 0.1813. This low reactivity is attributed to the relatively low content in ^{235}U (equivalent to UO_2 fuel with an enrichment of approximately up to 1.7 wt% ^{235}U), the large spacing between the rods (the pitch is approximately 1", the cladding OD is 0.412") and the absorption in the separator assembly. Together with the maximum k_{eff} values listed in Tables 6.1.7 and 6.1.8 this result demonstrates, that the k_{eff} for a Thoria Rod

Table 6.4.10

MAXIMUM k_{eff} VALUES WITH FILLED AND VOIDED GUIDE TUBES FOR THE MPC-32 AT 5.0 wt% ENRICHMENT

Fuel Class	Minimum Soluble Boron Content (ppm)	MPC-32 @ 5.0 %			
		Guide Tubes Filled,		Guide Tubes Voided,	
		1.0 g/cm^3	0.93 g/cm^3	1.0 g/cm^3	0.93 g/cm^3
14x14A	1900	0.8984	0.9000	0.8953	0.8943
14x14B	1900	0.9210	0.9214	0.9164	0.9118
14x14C	1900	0.9371	0.9376	0.9480	0.9421
14x14D	1900	0.9050	0.9027	0.8947	0.8904
14x14E	1900	0.7415	0.7301	n/a	n/a
15x15A	2500	0.9210	0.9223	0.9230	0.9210
15x15B	2500	0.9402	0.9420	0.9429	0.9421
15x15C	2500	0.9258	0.9292	0.9307	0.9293
15x15D	2600	0.9426	0.9419	0.9466	0.9440
15x15E	2600	0.9394	0.9415	0.9434	0.9442
15x15F	2600	0.9445	0.9465	0.9483	0.9460
15x15G	2500	0.9228	0.9244	0.9251	0.9243
15X15H	2600	0.9271	0.9301	0.9317	0.9333
15x15I‡	2500	0.9402	0.9363	-	-
16X16A	2000	0.9377	0.9375	0.9429	0.9389
16X16B	2000	0.9326	0.9338	0.9378	0.9358
16X16C	2000	0.9208	0.9193	0.9091	0.9055
17x17A	~~2600~~2200	0.9472~~0.9105~~	~~0.9145~~0.9468	~~0.9160~~0.9475	~~0.9161~~0.9459
17x17B	2600	0.9345	0.9358	0.9371	0.9356
17X17C	2600	0.9417	0.9431	0.9437	0.9430

‡ This array/class has solid guide rods that cannot be filled or voided.

Table 6.4.11

MAXIMUM k_{eff} VALUES WITH FILLED AND VOIDED GUIDE TUBES
FOR THE MPC-32 AT 4.1 wt% ENRICHMENT

Fuel Class	**Minimum Soluble Boron Content (ppm)**	**MPC-32 @ 4.1 %**			
		Guide Tubes Filled		**Guide Tubes Voided**	
		1.0 g/cm^3	**0.93 g/cm^3**	**1.0 g/cm^3**	**0.93 g/cm^3**
14x14A	1300	0.9041	0.9029	0.8954	0.8939
14x14B	1300	0.9257	0.9205	0.9128	0.9074
14x14C	1300	0.9402	0.9384	0.9423	0.9365
14x14D	1300	0.8970	0.8943	0.8836	0.8788
14x14E	1300	0.7340	0.7204	n/a	n/a
15x15A	1800	0.9199	0.9206	0.9193	0.9134
15x15B	1800	0.9397	0.9387	0.9385	0.9347
15x15C	1800	0.9266	0.9250	0.9264	0.9236
15x15D	1900	0.9375	0.9384	0.9380	0.9329
15x15E	1900	0.9348	0.9340	0.9365	0.9336
15x15F	1900	0.9411	0.9392	0.9400	0.9352
15x15G	1800	0.9147	0.9128	0.9125	0.9062
15X15H	1900	0.9267	0.9274	0.9276	0.9268
15x15I[§]	1800	0.9340	0.9316	-	-
16X16A	1400	0.9367	0.9347	0.9375	0.9308
16X16B	1400	0.9336	0.9319	0.9354	0.9283
16X16C	1400	0.9178	0.9130	0.9039	0.8965
17x17A	~~1900~~1600	~~0.9105~~0.9421	~~0.9111~~0.9413	~~0.9106~~0.9396	~~0.9091~~0.9357
17x17B	1900	0.9309	0.9307	0.9297	0.9243
17X17C	1900	0.9355	0.9347	0.9350	0.9308

[§] This array/class has solid guide rods that cannot be filled or voided.

Table 6.4.14

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-32
WITH UP TO 8 DFCs UNDER VARIOUS MODERATION CONDITIONS.

Fuel Assembly Class of Intact Fuel	Initial Enrichment (wt% ^{235}U)	Minimum Soluble Boron Content (ppm)	Maximum k_{eff}			
			Filled Guide Tubes		Voided Guide Tubes	
			1.0 g/cm^3	0.93 g/cm^3	1.0 g/cm^3	0.93 g/cm^3
14x14A through 14x14E	4.1	1500	0.9277	0.9283	0.9336	0.9298
	5.0	2300	0.9139	0.9180	0.9269	0.9262
15x15A, B, C, G, I	4.1	1900	0.9345	0.9350	0.9350	0.9326
	5.0	2700	0.9307	0.9346	0.9347	0.9365
15x15D, E, F, H	4.1	2100	0.9322	0.9336	0.9340	0.9329
	5.0	2900	0.9342	0.9375	0.9385	0.9397
16x16A, B, C	4.1	1500	0.9330	0.9332	0.9348	0.9333
	5.0	2300	0.9212	0.9246	0.9283	0.9299
17x17A	4.1	1800	0.9298	0.9310	0.9311	0.9283
	5.0	2600	0.9228	0.9274	0.9273	0.9298
17x17~~A,~~ B, C	4.1	2100	0.9284	0.9290	0.9294	0.9285
	5.0	2900	0.9308	0.9338	0.9355	0.9367

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-32, 4.1% Enrichment, Bounding Cases					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A03	HI-STAR	0.9041	0.9001	0.0006	0.3185
B14x14B01	HI-STAR	0.9257	0.9216	0.0007	0.4049
14x14C01	HI-STAR	0.9423	0.9382	0.0007	0.4862
14x14D01	HI-STAR	0.8970	0.8931	0.0006	0.5474
14x14E02	HI-STAR	0.7340	0.7300	0.0006	0.6817
15x15A01	HI-STAR	0.9206	0.9167	0.0006	0.5072
B15x15B01	HI-STAR	0.9397	0.9358	0.0006	0.4566
B15x15C01	HI-STAR	0.9266	0.9227	0.0006	0.4167
15x15D04	HI-STAR	0.9384	0.9345	0.0006	0.5594
15x15E01	HI-STAR	0.9365	0.9326	0.0006	0.5403
15x15F01	HI-STORM (DRY)	0.4691	0.4658	0.0003	1.207E+04
15x15F01	HI-TRAC	0.9403	0.9364	0.0006	0.4938
15x15F01	HI-STAR	0.9411	0.9371	0.0006	0.4923
15x15G01	HI-STAR	0.9147	0.9108	0.0006	0.5880
15x15H01	HI-STAR	0.9276	0.9237	0.0006	0.4710
15x15I01	HI-STAR	0.9340	0.9301	0.0006	0.5488
16x16A03	HI-STAR	0.9375	0.9333	0.0007	0.4488
16x16B01	HI-STAR	0.9354	0.9315	0.0006	0.4253
16x16C01	HI-STAR	0.9178	0.9137	0.0007	0.4408
17x17A01	HI-STAR	0.9421~~0.9111~~	0.9381~~0.9072~~	0.0006~~0.0006~~	0.3534~~0.4055~~
17x17B06	HI-STAR	0.9309	0.9269	0.0006	0.4365

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-32, 5.0% Enrichment, Bounding Cases					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
16x16A03	HI-STAR	0.9429	0.9388	0.0007	0.5920
16x16B01	HI-STAR	0.9378	0.9339	0.0006	0.5632
16x16C01	HI-STAR	0.9208	0.9167	0.0007	0.5898
17x17A01	HI-STAR	0.9475~~0.9161~~	0.9435~~0.9122~~	0.0006~~0.0006~~	0.5285~~0.6141~~
17x17B06	HI-STAR	0.9371	0.9331	0.0006	0.6705
17x17C02	HI-TRAC	0.9436	0.9396	0.0006	0.6773
17x17C02	HI-STAR	0.9437	0.9399	0.0006	0.6780

Note: Maximum k_{eff} = Calculated k_{eff} + $K_c \times \sigma_c$ + Bias + σ_B
where:
K_c = 2.0
σ_c = Std. Dev. (1-sigma)
Bias = 0.0021
σ_B = 0.0006
See Subsection 6.4.3 for further explanation.

SUPPLEMENT 6.III[1]: CRITICALITY EVALUATION OF THE MPC-68M

6.III.1 DISCUSSION AND RESULTS

In conformance with the principles established in NUREG-1536 [6.III.1.1], 10CFR72.124 [6.III.1.2], and NUREG-0800 Section 9.1.2 [6.III.1.3], the results in this supplement demonstrate that the effective multiplication factor (k_{eff}) of the HI-STORM 100 System with the MPC-68M, 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.

Criticality safety of the HI-STORM 100 System with the MPC-68M depends on the following principal design parameters:

- The inherent geometry of the fuel basket design of the MPC-68M;
- The incorporation of spatially distributed B-10 isotope in the Metamic-HT fuel basket structure. Based on the tests for the neutron absorber content in Metamic-HT (see Appendix 1.III.A and Supplement 9.III), and consistent with the approach taken for Metamic (see Section 9.1.5.3.2), 90% of the minimum B-10 (B_4C) content is credited in the analysis. With a specified minimum B_4C content of 10 wt%, the concentration credited in the analysis is therefore 9 wt%.

The off-normal and accident conditions defined in Section 2.2 are applicable to the HI-STORM System using the MPC-68M. These accidents are considered in Supplement 11.III and 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.

Results of the design basis criticality safety calculations for a single internally flooded HI-TRAC transfer cask with full water reflection on all sides (limiting cases for the HI-STORM 100 System), loaded with undamaged ~~intact~~ fuel assemblies are listed in Table 6.III.1.1, conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.III.3), and including the calculational bias, uncertainties, and calculational statistics. Table 6.III.1.1 provides the information for undamaged fuel without known or suspected cladding defects larger than pinhole leaks or hairline cracks, while Table 6.III.1.4 provides information for low-enriched, channeled BWR undamaged fuel without known or suspected grossly breached fuel rods. In addition, a result for a single internally dry (no moderator) HI-STORM storage cask with full water reflection on all external surfaces of the overpack, including the annulus region between the MPC and overpack, is listed in Table 6.III.1.2 to confirm the low reactivity of the HI-STORM 100 System with an MPC-68M in storage. The maximum k_{eff} for an MPC-68M loaded with up to 16 DFCs is listed in Table 6.III.1.3.

[1] Evaluations and results presented in this chapter are supported by documented calculation package(s) [6.III.1.4].

TABLE 6.III.1.1

BOUNDING MAXIMUM keff VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68M (HI-TRAC 100)

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% U-235)	Maximum keff
7x7B	4.8	0.9243
8x8B	4.8	0.9294
8x8C	4.8	0.9302
8x8D	4.8	0.9307
8x8E	4.8	0.9211
8x8F	4.5	0.9245
9x9A	4.8	0.9341
9x9B	4.8	0.9330
9x9C	4.8	0.9254
9x9D	4.8	0.9254
9x9E/F	4.5	0.9254
9x9G	4.8	0.9211
10x10A	4.8	0.9360
10x10B	4.8	0.9353
10x10C	4.8	0.9321
10x10F	4.7	0.9356
10x10G	4.75~~6~~	~~0.9393~~0.9472

Note: The results presented in the table above have an additional bias of 0.0021 applied to the 10x10 fuel assembly classes to conservatively account for any potential distributed enrichment effects. See Section 6.III.2.

TABLE 6.III.1.2

REPRESENTATIVE k_{eff} VALUES FOR MPC-68M IN THE HI-STORM 100 OVERPACK

Fuel Assembly Class	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**	**Maximum k_{eff}**
10x10A	4.8	0.3754

TABLE 6.III.1.3

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-68M
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	0.9408
8x8F and ~~,~~9x9E/F ~~and 10x10G~~	4.0	~~0.9131~~0.9028
10x10F and 10x10G	4.6	~~0.9362~~0.9453

TABLE 6.III.1.4

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-68M
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	0.9269

Note: The results presented in Tables 6.III.1.2, ~~and~~ 6.III.1.3 and 6.III.1.4 above have an additional bias of 0.0021 applied to the 10x10 fuel assembly classes to conservatively account for any potential distributed enrichment effects. See Section 6.III.2.

6.III.2 SPENT FUEL LOADING

The BWR fuel assembly classes/arrays which are authorized for the MPC-68 are qualified for the MPC-68M, except for the 6x6A, 6x6B, 6x6C, 7x7A, 8x8A, 10x10D and 10x10E. Additionally, the MPC-68M is qualified for two new assembly classes, 10x10F and 10x10G. Information on those classes is provided in Supplement 2.III, Table 2.III.3. Table 2.1.4 in Chapter 2 provides the acceptable fuel characteristics for all other fuel array/class authorized for storage in the MPC-68M, however fuel with planar-average initial enrichments up to 4.8 wt% ^{235}U are authorized in the MPC-68M.

BWR assemblies are specified in the Table 2.1.4 and Table 2.III.3 with a maximum planar-average enrichment. The analyses presented in this chapter use a uniform enrichment, equal to the maximum planar-average. Analyses presented in Appendix 6.B for the MPC-68 show that this is a conservative approach, i.e. that a uniform enrichment bounds the planar-average enrichment in terms of the maximum k_{eff}. To confirm this for the higher enrichments analyzed here, additional calculations were performed for the assembly class 10x10A in the MPC-68M, and are presented in Table 6.III.2.1 in comparison with the results for the uniform enrichment. Since the maximum planar-average enrichment of 4.8 wt% ^{235}U is above the actual enrichments of those assemblies, actual (as-built) enrichment distributions are not available. Therefore, several bounding cases are analyzed. Note that since the maximum planar-average enrichment of 4.8 wt% ^{235}U is close to the maximum rod enrichment of 5.0 wt% ^{235}U, the potential enrichment variations within the cross section are somewhat limited. To maximize the differences in enrichment under these conditions, the analyzed cases assume that about 50% of the rods in the cross section are at an enrichment of 5.0 wt% ^{235}U, while the remainder of the rods are at an enrichment of about 4.6 wt%, resulting in an average of 4.8 wt%. Calculations are performed for cross sections where all full-length and part-length, or only all full-length rods are present. For each case, two conditions are analyzed that places the different enrichment in areas with different local fuel-to-water ratios. Specifically, one condition places the higher enriched rods in locations where they are more surrounded by other rods, whereas the other condition places them in locations where they are more surrounded by water, such as near the water-rods or the periphery of the assembly. The results in Table 6.III.2.1 indicate that there may be a potential positive reactivity effect (+0.0021) due to distributed enrichments. Therefore, additional studies with distributed enrichments were performed and are presented in Table 6.III.2.2. These include all cases from Appendix B (for 8x8 and 9x9 assembly types), now evaluated in the MPC-68M, and additional cases for the 10x10G which has the highest reactivity of all assembly classes. The cases from Appendix B show no statistically significant increase, and in most cases a decrease in reactivity as a result of the distributed enrichment. ~~However, the assembly class 10x10G also shows a slight increase for one of the cases (+0.0012). Note that the small positive reactivity effect for the two 10x10 assembly classes is likely due to the very conservative selection of enriched rod locations used in the study (in the study they were placed close to the periphery and water rods, locations that are unlikely for actual fuel assemblies but were selected for the study to cover unknown rod patterns).~~ Nevertheless, for conservatism an additional bias of 0.0021 is applied to the results for all 10x10 fuel assembly classes in Section 6.III.1, including the cases with damaged fuel and low enriched channeled fuel. Note that for the studies presented in the remainder of this supplement this bias is not included since those studies focus on reactivity differences rather than absolute values of k_{eff}.

TABLE 6.III.2.2

ADDITIONAL CALCULATIONS OF THE REACTIVITY EFFECT OF DISTRIBUTED ENRICHMENTS IN BWR FUEL IN THE MPC-68M

Assembly Class	Enrichment	Maximum k_{eff}	Description	Delta-k_{eff}
8x8C	4.8	0.8273	Average Enrichment	-0.0044
8x8C	4.8	0.8229	Distributed Enrichment	
8x8C	4.8	0.8876	Average Enrichment	-0.0040
8x8C	4.8	0.8836	Distributed Enrichment	
8x8D	4.8	0.8550	Average Enrichment	+0.0004
8x8D	4.8	0.8554	Distributed Enrichment	
8x8D	4.8	0.8774	Average Enrichment	-0.0017
8x8D	4.8	0.8757	Distributed Enrichment	
8x8D	4.8	0.8855	Average Enrichment	-0.0026
8x8D	4.8	0.8829	Distributed Enrichment	
9x9B	4.8	0.9103	Average Enrichment	-0.0023
9x9B	4.8	0.9080	Distributed Enrichment	
9x9D	4.8	0.8467	Average Enrichment	-0.0095
9x9D	4.8	0.8372	Distributed Enrichment	
8x8C	4.8	0.9023	Average Enrichment	-0.0025
8x8C	4.8	0.8998	Distributed Enrichment	
8x8C	4.8	0.9165	Average Enrichment	-0.0003
8x8C	4.8	0.9162	Distributed Enrichment	
10x10G	4.75~~4.6~~	0.9451~~0.9372~~	Average Enrichment	-0.0253~~-0.0233~~
10x10G	4.75~~4.6~~	0.9198~~0.9139~~	Distributed Enrichment	
10x10G	4.75~~4.6~~	0.9451~~0.9372~~	Average Enrichment	-0.0268~~-0.0285~~
10x10G	4.75~~4.6~~	0.9183~~0.9087~~	Distributed Enrichment	
10x10G	4.75~~4.6~~	0.9451~~0.9372~~	Average Enrichment	-0.0009~~+0.0012~~
10x10G	4.75~~4.6~~	0.9442~~0.9384~~	Distributed Enrichment	
10x10G	4.75~~4.6~~	0.9451~~0.9372~~	Average Enrichment	-0.0010~~-0.0014~~
10x10G	4.75~~4.6~~	0.9441~~0.9358~~	Distributed Enrichment	

TABLE 6.III.3.1

EVALUATION OF BASKET MANUFACTURING TOLERANCES FOR MPC-68M

Box I.D.	**Box Wall Thickness**	**Maximum keff**
10x10A, 4.8% Enrichment		
nominal (6.05")	nominal (0.40")	0.9263
nominal (6.05")	minimum (0.38")	0.9307
increased (6.07")	minimum (0.38")	0.9288
minimum (5.99")	minimum (0.38")	0.9334
minimum, including deformation (5.96")	minimum (0.38")	**0.9339**
7x7B, 4.8% Enrichment		
nominal (6.05")	nominal (0.40")	0.9154
nominal (6.05")	minimum (0.38")	0.9196
minimum, including deformation (5.96")	minimum (0.38")	**0.9243**
8x8D, 4.8% Enrichment		
nominal (6.05")	nominal (0.40")	0.9230
nominal (6.05")	minimum (0.38")	0.9265
minimum, including deformation (5.96")	minimum (0.38")	**0.9307**
9x9A, 4.8% Enrichment		
nominal (6.05")	nominal (0.40")	0.9263
nominal (6.05")	minimum (0.38")	0.9301
minimum, including deformation (5.96")	minimum (0.38")	**0.9341**
10x10G, 4.75~~6~~% Enrichment		
nominal (6.05")	nominal (0.40")	0.9365~~0.9314~~
nominal (6.05")	minimum (0.38")	0.9409~~0.9349~~
minimum, including deformation (5.96")	minimum (0.38")	**0.9451~~0.9372~~**
10x10A, 4.8% Enrichment, Damaged Fuel		
nominal (6.05")	nominal (0.40")	0.9316
nominal (6.05")	minimum (0.38")	0.9348
minimum, including deformation (5.96")	minimum (0.38")	**0.9387**

Note: The results for the 10x10 fuel assembly classes do not include the bias for distributed enrichments discussed in Section 6.III.2.

6.III.4 CRITICALITY CALCULATIONS

The calculations in this supplement use the same computer codes and methodologies that are used in the main part of Chapter 6. Specifically, the conservative approach to model damaged fuel and fuel debris, using arrays of bare fuel rods, is the same (see discussion in Subsection 6.III.4.1 below).

The basket design of the MPC-68M is essentially identical to that of the MPC-68, in respect to the characteristics important to criticality safety. Specifically,

- The number and configuration of the cells for ~~intact~~undamaged and damaged fuel/fuel debris are unchanged;
- The basket dimensions are essentially the same; and
- The same poison material (B_4C) is used, but a larger ^{10}B content in the basket walls.

The content is also the same, except for the following

- Higher enrichments are qualified, consistent with the higher ^{10}B content in the basket walls; and
- Two additional fuel assembly types are analyzed, that are variations of existing types with slightly different dimension.

To verify that the bounding fuel parameter variations analyzed in the MPC-68 are also applicable to the MPC-68M, additional studies are performed and discussed in Subsection 6.III.4.2 below.

Due to the strong similarity in the basket design, the conclusions of the various studies presented in the main part of this Chapter on the MPC-68 are directly applicable to the MPC-68M. Nevertheless, to confirm this is also applicable to the MPC-68M, numerous studies with various moderation conditions that conclude that the fully flooded basket is the bounding case are re-analyzed and discussed in subsection 6.III.4.3. All analyzes are therefore performed under the following condition:

- Basket, and DFCs as applicable, are fully flooded with pure water at the maximum density; and
- Pellet-to-clad gaps of ~~intact~~undamaged assemblies are assumed flooded (see also discussion in Subsection 6.III.4.2 below)
- All assemblies and DFCs are located eccentrically in the basket, closest to the center of the basket.

Results for all design basis calculations are listed in Subsection 6.III.1. All maximum k_{eff} values are below the regulatory limit of 0.95.

6.III.4.1 Damaged Fuel and Fuel Debris

For damaged fuel and fuel debris in the MPC-68M the same conservative approach is used as in the main part of this chapter, see discussion in Section 6.4.4, specifically 6.4.4.2. Important aspects of this approach that ensure its conservatisms are as follows:

- All damaged fuel and fuel debris must be in damaged fuel containers (DFCs), and located in specifically designated cells on the periphery of the basket as specified in Table 2.1.22.
- A conservative model is used that bounds both damaged fuel and fuel debris. In other words, damaged fuel is always conservatively modeled as fuel debris.
- The model consists of regular arrays of fuel rods without cladding. The rods pitch (array size) is varied to determine the optimum moderation condition.
- ~~Intact~~Undamaged and damaged fuel/fuel debris in the same basket have the same enrichment limit, which may be different from the enrichment limit for ~~intact~~undamaged fuel only.
- The results for loading with ~~intact~~undamaged fuel only in Table 6.III.1.1 utilize different enrichment limits for different assembly classes, to ensure that the maximum k_{eff} is always below 0.95. It is therefore not possible to establish a single bounding assembly class/enrichment combination to be used in all analyzes with damaged fuel/fuel debris. Therefore, and in order to optimize the enrichment for the loading of ~~intact~~undamaged and damaged fuel/fuel debris for each assembly class, ~~intact~~undamaged assemblies are grouped by enrichment limit, and the ~~intact~~undamaged assembly with the highest maximum k_{eff} in each group is used for the calculations together with damaged fuel/fuel debris. These are:
  - ~~Intact~~Undamaged assemblies of 4.5 ~~and 4.6~~ wt%: Assembly class ~~10x10G~~9x9E/F. For the calculations with ~~intact~~undamaged and damaged fuel, an enrichment of 4.0 wt% is used.
  - ~~Intact~~Undamaged assembly of 4.7 and 4.75 wt%: Assembly class 10x10G~~F~~. For the calculations with ~~intact~~undamaged and damaged fuel, an enrichment of 4.6 wt% is used.
  - ~~Intact~~Undamaged assembly of 4.8 wt%: Assembly class 10x10A. For the calculations with ~~intact~~undamaged and damaged fuel, an enrichment of 4.8 wt% is used.
- Consistent with the results in the main part of this chapter for the MPC-68, array sizes of 9x9, 10x10 and 11x11 show the optimum moderation condition. This is confirmed for ~~intact~~undamaged assembly classes 9x9E/F, 10x10A and 10x10G by evaluating all arrays from 3x3 to 17x17 rods. ~~For assembly class 10x10F it is only confirmed that it is bounded by the cases with the 10x10A class (see Table 6.III.4.1).~~

6.III.4.2 Fuel Parameters and Parameter Variations

In the main part of the FSAR, extensive analyses of fuel dimensional variations have been performed. These calculations demonstrate that the maximum reactivity corresponds to:

- maximum active fuel length,
- maximum fuel pellet diameter,
- maximum fuel rod pitch,
- minimum cladding outside diameter (OD),
- maximum cladding inside diameter (ID),
- minimum guide tube/water rod thickness, and
- maximum channel thickness (for BWR assemblies only)
- part length rods (if present) removed.

The reason that those are bounding dimensions, i.e. that they result in maximum reactivity is directly based on, and can be directly derived from the three main characteristics affecting reactivity, namely 1) characteristics of the fission process; 2) the characteristics of the fuel assemblies and 3) the characteristics of the neutron absorber in the basket. These affect the reactivity as follows:

- The neutrons generated by fission are fast neutrons while the neutrons that initiate the fission need to be thermal neutrons. A moderator (water) is therefore necessary for the nuclear chain reaction to continue.
- Fuel assemblies are predominantly characterized by the amount of fuel and the fuel-to-water (moderator) ratio. Increasing the amount of fuel, or the enrichment of the fuel, will increase the amount of fissile material, and therefore increase reactivity. Regarding the fuel-to-water ratio, it is important to note that commercial BWR assemblies are undermoderated, i.e. they do not contain enough water for a maximum possible reactivity.
- The neutron poison in the basket walls uses B-10, which is an absorber of thermal neutrons. This poison therefore also needs water (moderator) to be effective. This places a specific importance on the amount of water between the outer rows of the fuel assemblies and the basket cell walls. Note that this explains some of the differences in reactivity between the different assembly types in the same basket, even for the same enrichment, where assemblies with a smaller cross section, i.e. which have more water between the periphery of the assembly and the surrounding wall, generally have a lower reactivity.

Based on these characteristics, the following conclusions can be made:

- Since fuel assemblies are undermoderated, any changes in geometry inside the fuel assembly that increases the amount of water while maintaining the amount of fuel are expected to increase reactivity. This explains why reducing the cladding or guide tube/water rod thicknesses, or increasing the fuel rod pitch results in an increase in reactivity.
- Increasing the active length will increase the amount of fuel while maintaining the fuel-to-water ratio, and therefore increase reactivity.
- The channel of the BWR assembly is a structure located outside of the rod array. It therefore does not affect the water-to-fuel ratio within the assembly. However, it reduces the amount of water between the assembly and the neutron poison, therefore reducing the effective thermalization for the poison. Therefore, an increase of the channel wall thickness will increase reactivity.
- In respect to the effect of the fuel pellet diameter, several compensatory effects need to be considered. Increasing the diameter will tend to increase the reactivity due to the increase in the fuel amount. However, it will also change the fuel-to-water-ratio, and will therefore make the fuel more undermoderated, which in turn tends to reduce reactivity. The effect of this change in moderation may depend on the condition of the pellet-to-clad gap. Assuming an empty pellet-to clad gap, which would be consistent with undamaged fuel rods, the change in moderation is small, and the net effect is an increase in reactivity, since the effect of the increase in the fissionable material dominates. In this case, the maximum pellet diameter is more reactive. When the pellet-to-clad gap is conservatively flooded, as recommended by NUREG 1536, a reduction of the fuel pellet diameter will also result in an increase in the amount of water, i.e. have a double effect on the water-to-fuel ratio. In this case, it is possible that a slight reduction may result in no reduction or even an increase in

reactivity. However, this is caused by a further amplification of the conservative assumption of the flooded pellet-to-clad gap, not by a positive increase in reactivity from the reduction in fuel (which would be counter-intuitive). Therefore, the maximum fuel pellet diameter is used for the fuel specification.

- Several assembly types contain part length rods (9x9A, 10x10A, B and G):
  - For 9x9A and 10x10A and B it was shown in the main part of the chapter (Tables 6.2.29, 36 and 37, respectively) that the condition with the part lengths rods completely removed is bounding. This condition is therefore used in the design basis calculations, so a specification of the lengths of the part length rods is not required. Applicability to the MPC-68M is confirmed by showing that all assemblies are undermoderated, which means the increase of water from completely removing the part length rods increases reactivity. All calculations for the MPC-68M for assembly classes 9x9A and 10x10A and B are therefore performed with models where the part length rods are replaced by water.
  - For assembly class 10x10G, the part length rods are located near the periphery and the water rod of the assembly, i.e. not surrounded by other rods on all sides. For this fuel assembly class, calculations with various part length rod lengths were performed and show that using full length rods in place of part length rods is more conservative. These calculations are listed in Table 6.III.4.4 ~~(Note that the 75% case and the 100% case in the tables are statistically equivalent, i.e. they do not indicate that a reduced length would result in a higher reactivity)~~. The case with all rods full length is therefore conservatively used in Table 6.III.1.1. This again removes the need to specify a length for the part lengths rods.

Since all assemblies have the same principal design, i.e. consist of bundles of cladded fuel rods, most of them with embedded guide/instrument tubes or water rods or channels, the above conclusions apply to all of them, and the bounding dimensions are therefore also common to all fuel assemblies analyzed here. Nevertheless, to clearly demonstrate that the main assumption is true, i.e. that all assemblies are undermoderated, a study was performed for all assembly types where the pellet-to-clad gap is empty instead of being flooded (a conservative assumption for the design basis calculations) The results are listed in Table 6.III.4.2, in comparison with the results of the reference cases with the flooded gap. In all cases, the reactivity is reduced compared to the reference case. This verifies that all assembly types considered here are in fact undermoderated, and therefore validates the main assumption stated above. All assembly types are therefore behaving in a similar fashion, and the bounding dimensions are therefore applicable to all assembly types.

The discussion provided above regarding the principal characteristics of fuel poison is also important for the various studies presented in this section, and supports the fact that those studies only need to be performed for a single BWR assembly type, and that the results of those studies are then generally applicable to all assembly types. The studies and the relationship to the discussion above are listed below.

Basket Manufacturing Tolerance: The two aspects of the basket tolerance that are evaluated are the cell wall thickness and the cell ID. The reduced cell wall thickness results in a reduced amount of poison (since the material composition of the wall is fixed), and therefore in an increase in

design with external moderators of various densities are shown in Table 6.III.4.7, all performed for the HI-TRAC 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. In a definitive study, Cano, et al. [6.4.2] 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. All calculations are therefore performed with full water density inside the MPCs.

- Partial Flooding: 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. 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.III.4.8. In all cases, the reactivity increases monotonically as the water level rises, confirming that the most reactive condition is fully flooded. Note that the studies for partial flooding are performed with the design basis model for the assembly class 10x10A that has the partial length rods removed for added conservatism, while the calculations in the main part of the chapter for the MPC-68 were performed for an assembly class that did not include partial length rods. This shows that the conclusion from partial flooding, i.e. that the fully flooded condition is bounding, applies equally to assemblies with and without partial lengths rods.

- Pellet-to-clad Gap Flooding: As demonstrated by the studies shown in Table 6.III.4.2, 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: 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 and therefore the fully flooded condition is bounding of the preferential flooding condition.

6.III.4.4 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 the storage locations of the MPC-68M 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 (Copper Induced Localized Corrosion) 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, 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.

Calculations are performed with these assemblies in all cells of the MPC-68M, without DFCs. The results of this conservative analysis are listed in Table 6.III.4.9 and show that the system remains below the regulatory limit.

In addition, calculations are performed for the MPC-68M with checkerboard configuration of normal undamaged fuel and low enriched, channeled BWR fuel fuel without DFCs. The results of this analysis are listed in Table 6.III.4.10 and show that the reactivity remains below the regulatory limit and bounded by the reference undamaged fuel assembly in all cells.

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 is structurally sound it can be considered undamaged when loaded in an MPC-68M.

6.III.4.5 Thoria Rod Canister

The criticality evaluation of thoria rod canister was performed for MPC-68 or MPC-68F and results presented in Section 6.4.6 show that it is permissible to load the Thoria Rod Canister together with any approved content in a MPC-68 or MPC-68F. While only a single canister is qualified for storage, the analysis assumes such a canister in every basket cell, and calculates a very low reactivity of less than 0.2 for this condition, based on a UO2 content of 1.8 wt%. Since the MPC-68M has equal or better criticality performance than the MPC-68 due to the basket itself being made from the neutron absorber, Metamic-HT. Without any further evaluations it can therefore be concluded that, from a criticality perspective, the thoria rods with the actual composition can be safely stored in the HI-STORM 100 system in an MPC-68M canister.

TABLE 6.III.4.1

MAXIMUM k_{eff} VALUES IN THE MPC-68M WITH INTACT (UNDAMAGED) AND DAMAGED FUEL/FUEL DEBRIS

	Maximum k_{eff},		
Bare Rod Array inside the DFC	**Assembly Classes 8x8F and, 9x9E/F and 10x10G (4.0 wt%)**	**Assembly Class 10x10F and 10x10G (4.6 wt%)**	**All other assembly classes (4.8 wt%)**
3x3	0.89260.8985	0.9310n/c‡	0.9267
6x6	0.89420.9032	0.9338n/c	0.9295
8x8	0.89860.9070	0.9395n/c	0.9344
9x9	0.90280.9087	0.9414n/c	0.9371
10x10	0.90240.9110	0.9432n/c	0.9387
11x11	0.90240.9105	0.94200.9341	0.9381
12x12	0.90180.9099	0.9412n/c	0.9373
13x13	0.90070.9084	0.9397n/c	0.9353
14x14	0.89930.9075	0.9385n/c	0.9352
16x16	0.89850.9064	0.9376n/c	0.9335
17x17	0.89760.9042	0.9366n/c	0.9328

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

‡ n/c = not calculated

Table 6.III.4.2

MAXIMUM k_{eff} VALUES IN THE MPC-68M FOR VARIOUS FUEL TYPES WITH VOIDED PELLET TO CLAD GAP

Assembly Classes	Enrichment	Maximum k_{eff} (Voided Gap)	Reference k_{eff} (Flooded Gap)	Delta k_{eff}
7X7B	4.8	0.9185	0.9243	-0.0058
8x8B	4.8	0.9210	0.9294	-0.0084
8x8C	4.8	0.9243	0.9302	-0.0059
8x8D	4.8	0.9245	0.9307	-0.0062
8x8E	4.8	0.9152	0.9211	-0.0059
8x8F	4.5	0.9191	0.9245	-0.0054
9x9A	4.8	0.9290	0.9341	-0.0051
9x9B	4.8	0.9202	0.9330	-0.0128
9x9C	4.8	0.9203	0.9254	-0.0051
9x9D	4.8	0.9210	0.9254	-0.0044
9x9E	4.5	0.9157	0.9254	-0.0097
9x9G	4.8	0.9160	0.9211	-0.0051
10x10A	4.8	0.9311	0.9339	-0.0028
10x10B	4.8	0.9242	0.9332	-0.0090
10x10C	4.8	0.9253	0.9300	-0.0047
10x10F	4.7	0.9301	0.9335	-0.0034
10x10G	4.75~~4.6~~	0.9403~~0.9335~~	0.9451~~0.9372~~	-0.0048~~-0.0037~~

Note: The results for the 10x10 fuel assembly classes do not include the bias for distributed enrichments discussed in Section 6.III.2.

Table 6.III.4.4

MAXIMUM k_{eff} VALUES IN THE MPC-68M
FOR VARIOUS PART LENGTH ROD LENGTHS (10x10G, 4.756% Enrichment)

Maximum k_{eff}	Description
0.91930.9117	Full Length Rods Only
0.92790.9217	Part Length Rods 25% length
0.93770.9312	Part Length Rods 50% length
0.94390.9374	Part Length Rods 75% length
0.94510.9372	All Rods

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

Table 6.III.4.7

MAXIMUM k_{eff} VALUES IN THE MPC-68M
FOR EXTERNAL FLOODING

Internal Water Density (%)	External Water Density (%)	7x7B (4.8%)	8x8F (4.5%)	9x9C (4.8%)	10x10A (4.8%)	10x10G (4.75~~6~~%)
100	100	0.9243	0.9245	0.9254	0.9351	0.9451~~0.9372~~
100	70	0.9238	0.9250	0.9259	0.9353	0.9450~~0.9388~~
100	50	0.9235	0.9239	0.9249	0.9336	0.9456~~0.9380~~
100	20	0.9234	0.9245	0.9259	0.9342	0.9452~~0.9383~~
100	10	0.9234	0.9245	0.9257	0.9351	0.9446~~0.9390~~
100	05	0.9238	0.9247	0.9258	0.9346	0.9458~~0.9387~~
100	01	0.9230	0.9256	0.9261	0.9341	0.9459~~0.9377~~

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

TABLE 6.III.4.9

MAXIMUM k_{eff} VALUES IN THE MPC-68M WITH LOW ENRICHED (3.3 wt% ^{235}U), CHANNELED BWR FUEL IN ALL CELLS

Rod Array inside the Channel	Maximum k_{eff}
3x3	0.2045
6x6	0.7229
8x8	0.8900
9x9	0.9219
10x10	0.9248
11x11	0.9065
12x12	0.8689
13x13	0.8161
14x14	0.7562
16x16	0.6653
17x17	0.6449

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

Table 6.III.4.10

MAXIMUM k_{eff} VALUES IN THE MPC-68M WITH MIXTURE OF UNDAMAGED BWR FUEL AND LOW ENRICHED (3.3 wt% ^{235}U), CHANNELED BWR FUEL

Configuration	Rod Array	10x10A, 4.8 wt% ^{235}U		10x10G, 4.75 wt% ^{235}U	
		Maximum k_{eff}	Reactivity Effect	Maximum k_{eff}	Reactivity Effect
Undamaged Normal Fuel in all Cells	-	0.9339	Reference	0.9451	Reference
Checkerboard of CILC Fuel at 3.3 wt% ^{235}U and Undamaged Fuel	3x3	0.6218	-0.3121	0.6247	-0.3204
	6x6	0.8241	-0.1098	0.8281	-0.1170
	8x8	0.9110	-0.0229	0.9161	-0.0290
	9x9	0.9275	-0.0064	0.9329	-0.0122
	10x10	0.9297	-0.0042	0.9341	-0.0110
	11x11	0.9206	-0.0133	0.9264	-0.0187
	12x12	0.9054	-0.0285	0.9109	-0.0342
	13x13	0.8865	-0.0474	0.8913	-0.0538
	14x14	0.8666	-0.0673	0.8719	-0.0732
	16x16	0.8514	-0.0825	0.8563	-0.0888
	17x17	0.8505	-0.0834	0.8561	-0.0890

Note: The results do not include the bias for distributed enrichments discussed in Section 6.III.2.

6.III.5 CRITICALITY BENCHMARK EXPERIMENTS

Same as in Section 6.5

6.III.6 REGULATORY COMPLIANCE

Same as in Section 6.6

6.III.7 REFERENCES

[6.III.1.1] NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, USNRC, Washington, D.C., January 1997.

[6.III.1.2] 10CFR72.124, "Criteria for Nuclear Criticality Safety."

[6.III.1.3] USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 2 - July 1981.

[6.III.1.4] "HI-STAR 100 AND HI-STORM 100 ADDITIONAL CRITICALITY CALCULATIONS", Holtec Report HI-2012771 Rev.20~~15~~ (proprietary)

8.2 ISFSI OPERATIONS

The HI-STORM 100 System heat removal system is a totally passive system. Maintenance on the HI-STORM system is typically limited to cleaning and touch-up painting of the overpacks, repair and replacement of damaged vent screens, and removal of vent blockages (e.g., leaves, debris). The heat removal system operability surveillance should be performed after any event that may have an impact on the safe functioning of the HI-STORM system. These include, but are not limited to, wind storms, heavy snow storms, fires inside the ISFSI, seismic activity, flooding of the ISFSI, and/or observed animal or insect infestations. The responses to these conditions involve first assessing the dose impact to perform the corrective action (inspect the HI-STORM overpack, clear the debris, check the cask pitch, and/or replace damaged vent screens), perform the corrective action, verify that the system is operable (check ventilation flow paths and radiation). In the event of significant damage to the HI-STORM, the situation may warrant removal of the MPC, and repair or replacement of the damaged HI-STORM overpack. If necessary, the procedures in Section 8.1 may be used to reposition a HI-STORM overpack for minor repairs and maintenance. In extreme cases, Section 8.3 may be used as guidance for unloading the MPC from the HI-STORM.

Note:
The heat removal system operability surveillance involves performing a visual examination on the HI-STORM exit and inlet vent screens to ensure that the vents remain clear or verifying the temperature rise from ambient to outlet is within prescribed limits. The metallic vent screens if damaged may allow leaves, debris or animals to enter the duct and block the flow of air to the MPC.

ALARA Warning:
Operators should practice ALARA principals when inspecting the vent screens. In most cases, binoculars allow the operator to perform the surveillance from a low dose area.

8.2.1 Perform the heat removal operability surveillance.

Note:
CAUTION: Some LCO requirements in the HI-STORM 100 CoC are based on individual system parameters (such as MPC total heat load). Sites should be aware of the variation in these requirements, and ensure procedures clearly identify how to implement these variations.

8.2.2 ISFSI Security Operations shall be performed in accordance with the approved site security program plan.

from the bottom of the overpack, the minimum distance to the controlled area boundary, was calculated for the HI-STORM 100S Version B with an MPC-24 for an assumed accident duration of 30 days. The burnup and cooling time of the fuel was 60,000 MWD/MTU and 3 years, which is more conservative than consistent with the off-site dose analysis reported in Chapter 1~~0~~1, Table 1~~0~~1.4.1 and the . This combination of overpack, MPC, burnup and cooling time is the same as that used in Chapters 5 and 10 for off-site dose calculations. The results presented below demonstrate that the regulatory requirements of 10CFR72.106 are easily met.

Distance	Dose Rate (mrem/hr)	Accident Duration	Total Dose (mrem)	10CFR72.106 Limit (mrem)
100 meters	2.36	720 hours or 30 days	1699.2	5000

11.2.3.4 Tip-Over Accident Corrective Action

Following a tip-over accident, the ISFSI operator shall first perform a radiological and visual inspection to determine the extent of the damage to the overpack. Special handling procedures, including the use of temporary shielding, will be developed and approved by the ISFSI operator.

If upon inspection of the MPC, structural damage of the MPC is observed, the structural damage shall be assessed and a determination shall be made if repairs will enable the MPC to return to service. If determined necessary, the MPC shall be returned to the facility for fuel unloading or transferred to either a HI-STAR or HI-STORM overpack in accordance with Chapter 8 for a duration that is determined to be appropriate. Likewise, the HI-STORM overpack shall be thoroughly inspected and a determination shall be made if repairs are required and will enable the HI-STORM overpack to return to service. Subsequent to the repairs, the equipment shall be inspected and appropriate tests shall be performed to certify the HI-STORM 100 System for service. If the equipment cannot be repaired and returned to service, the equipment shall be disposed of in accordance with the appropriate regulations.

11.2.4 Fire Accident

11.2.4.1 Cause of Fire

Although the probability of a fire accident affecting a HI-STORM 100 System during storage operations is low, a conservative fire has been assumed and analyzed. The analysis shows that the HI-STORM 100 System continues to perform its structural, confinement, thermal, and subcriticality functions.

11.2.13 100% Blockage of Air Inlets

11.2.13.1 Cause of 100% Blockage of Air Inlets

This event is defined as a complete blockage of all four bottom inlets. Such blockage of the inlets may be postulated to occur as a result of a flood, blizzard snow accumulation, tornado debris, or volcanic activity.

11.2.13.2 100% Blockage of Air Inlets Analysis

The immediate consequence of a complete blockage of the air inlet ducts 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 large mass, and correspondingly large thermal capacity of the storage overpack, it is expected that a significant temperature rise is only possible if the blocked condition is allowed to persist for a number of days. This accident condition is, however, a short duration event that will be identified and corrected by scheduled periodic surveillance at the ISFSI site depending on the cask heat load at the time of inspection. The temperature rise due to this accident event is small for heat loads much lower than design maximum heat load even if the condition persists for a number of days. As evaluated in Sub-section 4.6.2.4, mandatory 30-day surveillance of casks is required under heat loads less than or equal to the threshold heat load specified in Table 4.6.8 at the time of inspection.

Structural

There are no structural consequences as a result of this event.

Thermal

A thermal analysis is performed in Subsection 4.6.2 to determine the effect of a complete blockage of all inlets for an extended duration. For this event, both the fuel cladding and component temperatures remain below their short-term temperature limits. The MPC internal pressure for this event is evaluated in Subsection 4.6.2 and is bounded by the design basis internal pressure for accident conditions (Table 2.2.1).

Shielding

There is no effect on the shielding performance of the system as a result of this event, since the concrete temperatures do not exceed the short-term condition design temperature provided in Table 2.2.3.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the 100% blockage of air inlets accident does not affect the safe operation of the HI-STORM 100 System, if the blockage is removed in the specified time period.

11.2.13.3 100% Blockage of Air Inlets Dose Calculations

As shown in the analysis of the 100% blockage of air inlets accident, the shielding capabilities of the HI-STORM 100 System are unchanged because the peak concrete temperature does not exceed its short-term condition design temperature. The elevated temperatures will not cause the breach of the confinement system and the short term fuel cladding temperature limit is not exceeded. Therefore, there is no radiological impact.

11.2.13.4 100% Blockage of Air Inlets Accident Corrective Action

Analysis of the 100% blockage of air inlet accident shows that the temperatures for cask system components and fuel cladding are within the accident temperature limits if the blockage is cleared within 32 hours for cask heat loads greater than that specified in Table 4.6.8 at the time of inspection. For cask containing MPCs with heat loads less than or equal to that threshold heat load (Table 4.6.8), blockage is cleared within 31 days. Upon detection of the complete blockage of the air inlet ducts, the ISFSI operator shall assign personnel to clear the blockage with mechanical and manual means as necessary. After clearing the overpack ducts, the overpack shall be visually and radiologically inspected for any damage. If exit air temperature monitoring is performed in lieu of direct visual inspections, the difference between the ambient air temperature and the exit air temperature will be the basis for assurance that the temperature limits are not exceeded.

For an accident event that completely blocks the inlet or outlet air ducts of a cask with heat loads greater than the threshold heat load (Table 4.6.8) at the time of occurrence for greater than the analyzed duration, a site-specific evaluation or analysis may be performed to demonstrate adequate heat removal for the duration of the event. Adequate heat removal is defined as the minimum rate of heat dissipation that ensures cladding temperatures limits are met and structural integrity of the MPC and Overpack is not compromised. For those events where an evaluation or analysis is not performed or is not successful in showing that cladding temperatures remain below their short term

b. Canister material mechanical properties for structural integrity of the confinement boundary.

c. Canister and basket material thermal properties and dimensions for heat transfer control.

d. Canister and basket material composition and dimensions for dose rate control.

12.2.9 HI-STORM Overpack/VVM

a HI-STORM overpack/VVM material mechanical properties and dimensions for structural integrity to provide protection of the MPC and shielding of the spent nuclear fuel assemblies during loading, unloading and handling operations, as applicable.

b. HI-STORM overpack/VVM material thermal properties and dimensions for heat transfer control.

c. HI-STORM overpack/VVM material composition and dimensions for dose rate control.

12.2.10 Deleted~~Verifying Compliance with Fuel Assembly Decay Heat, Burnup, and Cooling Time Limits~~

~~The examples below execute the approach and equations described in Section 2.1.9.1 for determining allowable decay heat per storage location, burnup, and cooling time for the approved cask contents.~~

~~Example 1~~

~~In this example, a demonstration of the use of burnup versus cooling time tables for regionalized fuel loading is provided. In this example it will be assumed that the MPC-32 is being loaded with array/class 16x16A fuel in a regionalized loading pattern and will be stored in an aboveground HI-STORM system.~~

~~Step 1: Pick a value of X between 0.5 and 3. For this example X will be 2.8.~~

~~Step 2: Calculate $q_{Region2}$ as described in Section 2.1.9.1.2:~~

~~$q_{Region2} = (2 \times 34)/[(1 + (2.8)^{0.2075}) \times ((12 \times 2.8) + 20)] = 0.5668$ kW‡~~

~~Step 3: Calculate $q_{Region1}$ as described in Section 2.1.9.1.2:~~

~~$q_{Region1} = X \times q_{Region2} = 2.8 \times 0.5668 = 1.5871$ kW~~

~~‡ Results are arbitrarily rounded to four decimal places.~~

Step 4: Develop a burnup versus cooling time table. Since this table is enrichment dependent, it is permitted and advisable to create multiple tables for different enrichments. In this example, two enrichments will be used: 3.1 and 4.185. Tables 12.2.1 and 12.2.2 show the burnup versus cooling time tables calculated for these enrichments for Region 1 and Region 2 as described in Section 2.1.9.1.3.

Table 12.2.3 provides three hypothetical fuel assemblies in the 16x16A array/class that will be evaluated for acceptability for loading in the MPC-32 example above. The decay heat values in Table 12.2.3 are calculated by the user. The other information is taken from the fuel assembly and reactor operating records.

Fuel Assembly Number 1 is not acceptable for storage because its enrichment is lower than that used to determine the allowable burnups in Table 12.2.1 and 12.1.2. The solution is to develop another table using an enrichment of 3.0 wt.% ^{235}U or less to determine this fuel assembly's suitability for loading in this MPC-32.

Fuel Assembly Number 2 is not acceptable for loading unless a unique maximum allowable burnup for a cooling time of 3.3 years is calculated by linear interpolation between the values in Table 12.2.1 for 3 years and 4 years of cooling. Linear interpolation yields a maximum burnup of 36,497 MWD/MTU (rounded down from 36,497.2), making Fuel Assembly Number 2 acceptable for loading only in Region 1 due to decay heat limitations.

Fuel Assembly Number 3 is acceptable for loading based on the higher allowable burnups in Table 12.2.2, which were calculated using a higher minimum enrichment that those in Table 12.2.1, which is still below the actual initial enrichment of Fuel Assembly Number 3. Due to its relatively low total decay heat of 0.5 kW (fuel: 0.4, non-fuel hardware: 0.1), Fuel Assembly Number 3 may be stored in Region 1 or Region 2.

Example 2

In this example, each fuel assembly in Table 12.2.3 will be evaluated to determine whether it may be stored in the same hypothetical MPC-32 in a regionalized storage pattern in an aboveground system. Assuming the same value 'X', the same maximum fuel storage location decay heats are calculated. The equation in Section 2.1.9.1.3 is executed for each fuel assembly using its exact initial enrichment to determine its maximum allowable burnup. Linear interpolation is used to further refine the maximum allowable burnup value between cooling times, if necessary.

Fuel Assembly Number 1: The calculated allowable burnup for 3.0 wt.% ^{235}U and a decay heat value of 1.5871 kW ($q_{region1}$) is 44,905 MWD/MTU at 4 years minimum cooling. Its decay heat is too high for loading in Region 2. Comparing the fuel assembly burnup and total decay heat of the contents‡ (fuel (1.01 kW) plus non-fuel hardware (0.5 kW)) to the calculated limits indicates that the fuel assembly, including the non-fuel hardware, is acceptable for storage in Region 1.

‡ The assumption is made that the non-fuel hardware meets burnup and cooling time limits in Table 2.1.25.

~~Fuel Assembly Number 2: The calculated allowable burnup for 3.2 wt.% ^{235}U and a decay heat value of 1.5871 kW ($q_{region1}$) is 32,989 MWD/MTU for 3 years cooling and 45,382 MWD/MTU for 4 years cooling. Linearly interpolating between these values for a cooling time of 3.3 years yields a maximum allowable burnup of 36,706 MWD/MTU and, therefore, the assembly is acceptable for storage in Region 1. This fuel assembly's decay heat is also too high for loading in Region 2.~~

~~Fuel Assembly Number 3: The calculated allowable maximum burnup for 4.3 wt.% ^{235}U and a decay heat value of 0.5668 ($q_{Region2}$) is 41,693 MWD/MTU for 18 years cooling. Comparing the fuel assembly burnup and total decay heat of the contents (fuel plus non-fuel hardware) against the calculated limits indicates that the fuel assembly and non-fuel hardware are acceptable for storage. Therefore, the assembly is acceptable for storage in Region 2. This fuel assembly would also be acceptable for loading in Region 1 (this conclusion is inferred, but not demonstrated).~~

~~Example 3~~

~~In this example, a demonstration of the use of burnup versus cooling time tables for uniform fuel loading is provided. In this example it will be assumed that the MPC-68 is being loaded with array/class 9x9A fuel and will be stored in an aboveground HI-STORM system.~~

~~Step 1: CoC TS Appendix B Table 2.4-1 provides the heat load limit on each storage location (q_{max}). For MPC-68 this is 0.5 kW.~~

~~Step 2: Develop a burnup versus cooling time table. Since this table is enrichment dependent, it is permitted and advisable to create multiple tables for different enrichments if the fuel being loaded varies significantly in initial enrichment. It is conservative to choose the lowest value of initial enrichment to generate the table.~~

~~In this example, two enrichments will be used: 3.0 and 4.5. Tables 12.2.4 and 12.2.5 show the burnup versus cooling time tables calculated for these enrichments for the respective q_{max}.~~

~~Table 12.2.6 provides three hypothetical fuel assemblies in the 9x9A array/class that will be evaluated for acceptability for loading in the MPC-68 example above. The decay heat values in Table 12.2.6 would be calculated by the user. The other information would be taken from the fuel assembly and reactor operating records.~~

~~All of the assemblies meet the per cell heat load limit of 0.5 kW.~~

~~Fuel Assembly Number 1 is acceptable for storage because its enrichment is lower than that used to determine the allowable burnups in Table 12.2.4 and the burnup is lower than that allowed for the cooling time of the assembly.~~

~~Fuel Assembly Number 2 is not acceptable for loading based on the current tables. The fuel assembly burnup is greater than allowed by Table 12.2.4, even with linear interpolation (30978~~

~~MWD/MTU). Fuel Assembly Number 2 may be acceptable for loading if a new table is created specifically for an initial enrichment of 3.5 wt% and the allowable burnup is greater than 35250.~~

~~Fuel Assembly Number 3 is acceptable for loading based on the allowable burnups in Table 12.2.5.~~

12.2.11 Verifying Compliance with Total MPC Heat Load

Some operational steps and/or use of particular equipment are required if Q_{CoC} is above a certain value, e.g. 28.74 kW in the MPC-32. These include supplemental cooling, forced helium dehydration, helium backfill pressure, and surveillance requirements for LCO 3.1.2. These examples demonstrate the logic behind the decisions for these operational steps. Time to boil limits and vacuum drying are also considered in these examples.

Example 1:

Table 12.2.7 contains a proposed heat load pattern for loading a MPC-68 into an aboveground HI-STORM 100 System. The table provides the decay heat of each storage location. ~~It is assumed that each of these assemblies meets the burnup, cooling time and enrichment criteria for loading as described in the previous examples in Section 12.2.10.~~

General observations on this loading plan:

1. The heat loads in all cells meet the CoC limits for Uniform Loading, i.e. all cells are ≤ 0.50 kW (See Table 2.1.26).
2. The MPC is loaded preferentially for ALARA considerations, i.e. the assemblies with the lower heat loads are in the peripheral cells.
3. The aggregate MPC heat load, as defined in Section 2.1.9.1.2 as the simple summation of the assemblies in the MPC, is 18.917 kW.
4. The maximum heat load in any cell is 0.460 kW.
5. Q_{CoC}, as defined in Section 2.1.9.1.2 equation c is 31.280 kW.

Recommendations based on the general observations without further site-specific analysis:

1. Vacuum drying: The MPC *cannot* be dried using vacuum drying because the Q_{CoC} heat load is greater than 30 kW (See FSAR Table 4.5.1).
2. Forced Helium Dehydration: The MPC should be dried using forced helium dehydration since the Q_{CoC} heat load exceeds the vacuum drying threshold heat loads (See FSAR Table 4.5.1).
3. Helium Backfill Pressure Range: The MPC should be backfilled to the higher pressure range given in the TS because the Q_{CoC} heat load exceeds the threshold heat loads in FSAR Table 1.2.2.
4. Supplemental Cooling System: A supplemental cooling system would be required for on-site transport of High Burnup Fuel in the HI-TRAC after the MPC is dried, backfilled and sealed because the Q_{CoC} heat load exceeds the 90% design basis threshold heat load in FSAR Table 4.5.4.

5. Heat Removal Surveillance (LCO 3.1.2): The user has 24 hours to clear blockage on the system containing this MPC since the Q_{CoC} heat load exceeds the 28.152 kW (=0.414 kW*68) threshold heat load in LCO 3.1.2.
6. Time to boil determination: The user can calculate the time to boil limit based on the aggregate MPC heat load of 18.917 kW since this is a bulk adiabatic heat up calculation strictly based on the aggregate heat in the MPC.
7. Air mass flow rate test requirements per Condition 9 of the CoC: The user can determine if this test needs to be performed based on the aggregate MPC heat load of 18.917 kW since the air flow on the outside of the MPC is strictly based on the aggregate heat in the MPC.

Example 2

Table 12.2.8 contains a proposed heat load pattern for loading a MPC-32. The table provides the decay heat of each storage location. ~~It is assumed that each of these assemblies meets the burnup, cooling time and enrichment criteria for loading as described in the previous examples in Section 12.2.10.~~

General observations on this loading plan:

1. The heat loads in all cells meet the CoC limits for Uniform Loading, i.e. all cells are ≤ 1.062 kW (See Table 2.1.26).
2. The MPC is loaded preferentially for ALARA considerations, i.e. the assemblies with the lower heat loads are in the peripheral cells.
3. The aggregate MPC heat load, as defined in Section 2.1.9.1.2 as the simple summation of the assemblies in the MPC, is 17.471 kW.
4. The maximum heat load in any cell is 0.826 kW.
5. $Q_{CoC,}$ as defined in Section 2.1.9.1.2 equation c is 26.432 kW.

Recommendations based on the general observations without further site-specific analysis:

1. Vacuum drying: The MPC can be dried using vacuum drying since the Q_{CoC} heat load is bounded by the threshold heat load Q2 in FSAR Table 4.5.1. The vacuum drying is time limited as Q_{CoC} exceeds threshold heat load Q1 in FSAR Table 4.5.1.
2. Forced Helium Dehydration: The MPC can be dried using forced helium dehydration but it is not required.
3. Helium Backfill Pressure Range: The MPC may be backfilled to either pressure range given in the TS because the Q_{CoC} heat load is bounded by the threshold heat load in FSAR Table 1.2.2.
4. Supplemental Cooling System: A supplemental cooling system would NOT be required for on-site transport in the HI-TRAC after the MPC is dried, backfilled and sealed because the Q_{CoC} heat load is bounded by the 90% design basis threshold heat load in FSAR Table 4.5.4.
5. Heat Removal Surveillance (LCO 3.1.2): The user has 64 hours to clear blockage on

the system containing this MPC since the Q_{CoC} heat load is bounded by the 28.74 kW threshold heat load in LCO 3.1.2.

6. Time to boil determination: The user can calculate the time to boil limit based on the aggregate MPC heat load of 17.471 kW since this is a bulk adiabatic heat up calculation strictly based on the aggregate heat in the MPC.
7. Air mass flow rate test requirements per Condition 9 of the CoC: The user can determine if this test needs to be performed based on the aggregate MPC heat load of 17.471 kW since the air flow on the outside of the MPC is strictly based on the aggregate heat in the MPC.

Example 3

Table 12.2.9 contains a proposed heat load pattern for loading a MPC-32. The table provides the decay heat of each storage location. ~~It is assumed that each of these assemblies meets the burnup, cooling time and enrichment criteria for loading as described in the previous examples in Section 12.2.10.~~

General observations on this loading plan:

1. The heat loads do not meet the CoC limits for Uniform Loading, i.e. some cells are ≥ 1.0625 kW (See Table 2.1.26).
2. The X value that most closely meets this pattern (See Table 2.1.30) is 1.5 which means the inner locations cannot have a total decay heat greater than 1.282 kW and the outer locations cannot have a total decay heat greater than 0.855 kW. Note that the pattern also meets the criteria for any X value ≥ 1.5.
3. The aggregate MPC heat load, as defined in Section 2.1.9.1.2 as the simple summation of the assemblies in the MPC, is 20.697 kW.
4. The maximum heat load in any cell is 1.273 kW.
5. Since this MPC is loaded in a regionalized pattern, Q_{CoC}, as defined in Section 2.1.9.1.2 equation e is 32.484 kW. (12*1.282+20*0.855)

Recommendations based on the general observations <u>without further site-specific analysis</u>:

1. Vacuum drying: The MPC *cannot* be dried using vacuum drying since the Q_{CoC} heat load under uniform loading (1.273 kWx32 equals 40.736 kW) exceeds the threshold heat loads in FSAR Table 4.5.1.
2. Forced Helium Dehydration: The MPC must be dried using forced helium dehydration only because vacuum drying is not permitted (see above) and regionalized loading Q_{CoC} is bounded by the design basis heat load in FSAR Table 4.5.1.
3. Helium Backfill Pressure Range: The MPC must be backfilled to the higher pressure range given in the TS because the uniform loading Q_{CoC} heat load exceeds the threshold heat load in FSAR Table 1.2.2.
4. Supplemental Cooling System: A supplemental cooling system is required for on-site

Table 12.2.1

[INTENTIONALLY DELETED]

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING
(MPC-32, Array/Class 16x16A, X = 2.8, and Enrichment = 3.1 wt.% ^{235}U)
($q_{Region\ 1}$ = 1.5871 kW, $q_{Region\ 2}$ = 0.5668 kW)

MINIMUM COOLING TIME (years)	MAXIMUM ALLOWABLE BURNUP IN REGION 1 (MWD/MTU)	MAXIMUM ALLOWABLE BURNUP IN REGION 2 (MWD/MTU)
≥3	32791	10896
≥4	45145	17370
≥5	53769	22697
≥6	59699	26615
≥7	63971	29386
≥8	67343	31437
≥9	68200	33000
≥10	68200	34271
≥11	68200	35384
≥12	68200	36322
≥13	68200	37189
≥14	68200	37980
≥15	68200	38773
≥16	68200	39512
≥17	68200	40234
≥18	68200	40908
≥19	68200	41620
≥20	68200	42324

Table 12.2.2

[INTENTIONALLY DELETED]

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING
(MPC-32, Array/Class 16x16A, X = 2.8, and Enrichment =4.185 wt.% ^{235}U)
($q_{Region\ 1}$ = 1.5871 kW, $q_{Region\ 2}$ = 0.5668 kW)

MINIMUM COOLING TIME (years)	MAXIMUM ALLOWABLE BURNUP IN REGION 1 (MWD/MTU)	MAXIMUM ALLOWABLE BURNUP IN REGION 2 (MWD/MTU)
≥3	34797	11101
≥4	47590	17870
≥5	56438	23272
≥6	62533	27157
≥7	66963	29907
≥8	68200	31935
≥9	68200	33510
≥10	68200	34785
≥11	68200	35927
≥12	68200	36894
≥13	68200	37790
≥14	68200	38593
≥15	68200	39419
≥16	68200	40191
≥17	68200	40937
≥18	68200	41643
≥19	68200	42363
≥20	68200	43094

Table 12.2.3

[INTENTIONALLY DELETED]

SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE
(Array/Class 16x16A)

FUEL ASSEMBLY NUMBER	ENRICHMENT (wt. % ^{235}U)	FUEL ASSEMBLY BURNUP (MWD/MTU)	FUEL ASSEMBLY COOLING TIME (years)	FUEL ASSEMBLY DECAY HEAT (kW)	NON-FUEL HARDWARE STORED WITH ASSEMBLY	NFH DECAY HEAT (kW)
1	3.0	37100	4.7	1.01	BPRA	0.5
2	3.2	35250	3.3	1.45	NA	NA
3	4.3	41276	18.2	0.4	BPRA	0.1

Table 12.2.4

[INTENTIONALLY DELETED]

~~EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING~~
~~(MPC-68, Array/Class 9x9A, and Enrichment = 3.0 wt.% ^{235}U)~~
~~(q_{max} = 0.5 kW)~~

~~MINIMUM COOLING TIME (years)~~	~~MAXIMUM ALLOWABLE BURNUP (MWD/MTU)~~
~~≥3~~	~~27739~~
~~≥4~~	~~38536~~
~~≥5~~	~~46268~~
~~≥6~~	~~51583~~
~~≥7~~	~~55424~~
~~≥8~~	~~58303~~
~~≥9~~	~~60733~~
~~≥10~~	~~62798~~
~~≥11~~	~~64609~~
~~≥12~~	~~66331~~
~~≥13~~	~~68005~~
~~≥14~~	~~68200~~
~~≥15~~	~~68200~~
~~≥16~~	~~68200~~
~~≥17~~	~~68200~~
~~≥18~~	~~68200~~
~~≥19~~	~~68200~~
~~≥20~~	~~68200~~

Table 12.2.5

[INTENTIONALLY DELETED]

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING
(MPC-68, Array/Class 9x9A, and Enrichment =4.5 wt.% ^{235}U)
(q_{max} = 0.5 kW)

MINIMUM COOLING TIME (years)	MAXIMUM ALLOWABLE BURNUP (MWD/MTU)
≥3	30017
≥4	41399
≥5	49359
≥6	54839
≥7	58856
≥8	61932
≥9	64534
≥10	66802
≥11	68200
≥12	68200
≥13	68200
≥14	68200
≥15	68200
≥16	68200
≥17	68200
≥18	68200
≥19	68200
≥20	68200

Table 12.2.6

[INTENTIONALLY DELETED]

~~SAMPLE CONTENTS TO DETERMINE ACCEPTABILITY FOR STORAGE~~
~~(Array/Class 9x9A)~~

~~FUEL ASSEMBLY NUMBER~~	~~ENRICHMENT (wt. % ^{235}U)~~	~~FUEL ASSEMBLY BURNUP (MWD/MTU)~~	~~FUEL ASSEMBLY COOLING TIME (years)~~	~~FUEL ASSEMBLY DECAY HEAT (kW)~~
~~1~~	~~3.0~~	~~37100~~	~~4.7~~	~~0.3~~
~~2~~	~~3.5~~	~~35250~~	~~3.3~~	~~0.495~~
~~3~~	~~4.5~~	~~41276~~	~~18.2~~	~~0.2~~

BASES

LCO (continued)

This LCO is not intended to address low frequency, unexpected Design Event III and IV class events such as design basis accidents and extreme environmental phenomena that could potentially block one or more of the air ducts for an extended period of time (i.e., longer than the total Completion Time of the LCO). This class of events is addressed site-specifically as required by Section 3.4.9 of Appendix B to the CoC.

APPLICABILITY

The LCO is applicable during STORAGE OPERATIONS. Once an OVERPACK containing an MPC loaded with spent fuel has been placed into it's storage configuration, the heat removal system must be operable to ensure adequate dissipation of the decay heat from the fuel assemblies. In accordance with a safety first approach, when observed, blockage should be cleared from OVERPACK vents, even if the extent of blockage does not exceed the threshold that renders the system inoperable.

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each SFSC not meeting the LCO. Subsequent SFSCs that don't meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

~~Although the heat removal system remains operable, the blockage should be cleared expeditiously.~~Deleted

(continued)

ACTIONS (continued)

B.1

If the heat removal system has been determined to be inoperable, it must be restored to operable status within eight hours for OVERPACKS containing MPCs with heat loads exceeding 19 kW at the time of inspection. Eight hours is a reasonable period of time (typically, one operating shift) to take action to remove the obstructions in the air flow path.

Alternatively, for OVERPACKS containing MPCs with heat loads up to 19 kW at the time of inspection, the system must be restored to operable status within twenty four hours. Twenty four hours is a reasonable period of time for these lower heat load systems since the temperature limits of the system components and fuel cladding are not exceeded and the event is not time limiting.

(continued)

BASES

ACTIONS (continued)

C.1

For MPCs with heat loads greater than 19 kW, Iif the heat removal system cannot be restored to operable status within eight hours, the innermost portion of the OVERPACK concrete may experience elevated temperatures. Therefore, dose rates are required to be measured to verify the effectiveness of the radiation shielding provided by the concrete. This Action must be performed immediately and repeated every twelve hours thereafter to provide timely and continued evaluation of the effectiveness of the concrete shielding. As necessary, the cask user shall provide additional radiation protection measures such as temporary shielding. The Completion Time is reasonable considering the expected slow rate of deterioration, if any, of the concrete under elevated temperatures.
For MPCs with heat loads up to 19 kW, there will be inconsequential temperature increase to the OVERPACK concrete if the system is not restored to operable status within 24 hours. If the heat removal system cannot be restore to operable status within 24 hours, the same actions as above are required.

C.2.1

In addition to Required Action C.1, efforts must continue to restore cooling to the SFSC. Efforts must continue to restore the heat removal system to operable status by removing the air flow obstruction(s) unless optional Required Action C.2.2 is being implemented.

(continued)

ACTIONS

C.2.1 (continued)

This Required Action must be complete in 64 hours (after entering Condition C) for an aboveground system with an MPC decay heat load of 28.74 kW or less, in 24 hours (after entering Condition C) for an aboveground system with an MPC decay heat load greater than 28.74 kW, and in 16 hours for an underground system. These Completion Times are consistent with the thermal analyses of this event, which show that all component temperatures remain below their short-term temperature limits up to 72, 32 or 24 hours after event initiation, respectively. For MPC heat loads up to 19 kW, system components temperatures do not exceed their short term temperature limits.

The Completion Time reflects the 8 or 24 hours to complete Required Action B.1 and the appropriate balance of time consistent with the applicable analysis results. The event is assumed to begin at the time the SFSC heat removal system is declared inoperable. This is reasonable considering the low probability of all air ducts becoming simultaneously blocked by trash or debris.

C.2.2

In lieu of implementing Required Action C.2.1, transfer of the MPC into a TRANSFER CASK will place the MPC in an analyzed condition and ensure adequate fuel cooling until actions to correct the heat removal system inoperability can be completed. Transfer of the MPC into a TRANSFER CASK removes the SFSC from the LCO Applicability since STORAGE OPERATIONS does not include times when the MPC resides in the TRANSFER CASK. In this case, the requirements of CoC Appendix A, LCO 3.1.4 apply.

(continued)

ACTIONS

C.2.2 (continued)

An engineering evaluation must be performed to determine if any concrete deterioration has occurred which prevents it from performing its design function. If the evaluation is successful and the air flow obstructions have been cleared, the OVERPACK heat removal system may be considered operable and the MPC transferred back into the OVERPACK. Compliance with LCO 3.1.2 is then restored. If the evaluation is unsuccessful, the user must transfer the MPC into a different, fully qualified OVERPACK to resume STORAGE OPERATIONS and restore compliance with LCO 3.1.2

In lieu of performing the engineering evaluation, the user may opt to proceed directly to transferring the MPC into a different, fully qualified OVERPACK or place the TRANSFER CASK in the spent fuel pool and unload the MPC.

The Completion Times of 64 hours, 24 hours and 16 hours reflect the Completion Time from Required Action C.2.1 to ensure component temperatures remain below their short-term temperature limits for the respective decay heat loads and OVERPACK styles.

SURVEILLANCE REQUIREMENTS

SR 3.1.2

The long-term integrity of the stored fuel is dependent on the ability of the SFSC to reject heat from the MPC to the environment. There are two options for implementing SR 3.1.2, either of which is acceptable for demonstrating that the heat removal system is OPERABLE.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.2 (continued)

Visual observation that all air inlets and outlets are unobstructed ensures that air flow past the MPC is occurring and heat transfer is taking place. Greater than 50% blockage of the total air inlet area or air outlet area renders the heat removal system inoperable and this LCO is not met. 50% or less blockage of the total air inlet area or air outlet area does not constitute inoperability of the heat removal system. However, corrective actions should be taken as promptly as conditions permit to remove the obstruction and restore full flow through the affected air duct(s).

As an alternative, for OVERPACKs with air temperature monitoring instrumentation installed in the air outlets, the temperature rise between ambient and the OVERPACK air outlet may be monitored to verify operability of the heat removal system. Blocked air ducts will reduce air flow and increase the temperature rise experienced by the air as it removes heat from the MPC. Based on the analyses, provided the air temperature rise is less than the limit stated in the SR, adequate air flow and, therefore, adequate heat transfer is occurring to provide assurance of long term fuel cladding integrity. The reference ambient temperature used to perform this Surveillance shall be measured at the ISFSI facility.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2 (continued)

The Frequency of 24 hours for aboveground systems with heat loads that exceed 19 kW at the time of inspection, and 16 hours for underground systems is reasonable based on the time necessary for SFSC components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of blockage of air ducts. For aboveground systems containing MPCs with heat loads up to 19 kW at the time of inspection, the surveillance frequency of 30 days is appropriate, since the system components and peak cladding temperature limits for accident conditions are not exceeded and the event is not time limiting.

REFERENCES

1. FSAR Chapter 4
2. FSAR Sections 11.2.13 and 11.2.14
3. ANSI/ANS 57.9-1992